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10 CFR 52.79

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

LEVY NUCLEAR PLANT, UNITS 1 AND 2
DOCKET NOS. 52-029 AND 52-030
PARTIAL RESPONSE TO NRC RAI LETTERS 116, 117 and 118 – SRP SECTIONS 6.3 AND 15.2.6

References:
1. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated March 6, 2014, "Request for Additional Information Letter No. 116 Related to SRP Sections 6.3 and 15.2.6."
2. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated April 10, 2014, "Request for Additional Information Letter No. 117 Related to SRP Section 6.3."
3. Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated April 24, 2014, "Request for Additional Information Letter No. 118 Related to SRP Section 6.3."
4. Letter from Christopher M. Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated April 17, 2014, "Partial Response to NRC RAI Letter 116 – SRP Sections 6.3 and 15.2.6," Serial: NPD-NRC-2014-012
5. Letter from Christopher M. Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated May 5, 2014, "Partial Response to NRC RAI Letter 116 – SRP Sections 6.3 and 15.2.6," Serial: NPD-NRC-2014-014
6. Letter from Christopher M. Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated May 19, 2014, "Partial Response to NRC RAI Letter 116 – SRP Sections 6.3 and 15.2.6," Serial: NPD-NRC-2014-015
7. Letter from Christopher M. Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated June 12, 2014, "Partial Response to NRC RAI Letter 116 – SRP Sections 6.3 and 15.2.6," Serial: NPD-NRC-2014-016
8. Letter from Christopher M. Fallon (DEF) to Nuclear Regulatory Commission (NRC), dated June 19, 2014, "Partial Response to NRC RAI Letter 116 – SRP Sections 6.3 and 15.2.6," Serial: NPD-NRC-2014-017

Ladies and Gentlemen:

Duke Energy Florida, Inc. (DEF) hereby submits a partial response to the Nuclear Regulatory Commission's (NRC) request for additional information (RAI) cited in References 1, 2 and 3.

Enclosure 1 to this letter contains DEF's partial response consisting of responses to RAI Questions15.02.06-2, 06.03-5, 06.03-10, 06.03-11 and 06.03-12. Attachment A to Enclosure 1 contains the proprietary version of the response to RAI Questions 15.02.06-2 and 06.03-5 and Attachment B to Enclosure 1 contains a redacted, non-proprietary version of the response to RAI Questions 15.02.06-2 and 06.03-5. Responses to questions provided previously are contained in References 4, 5, 6, 7 and 8.

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As Attachment A to Enclosure 1 contains information proprietary to Westinghouse Electric Company, LLC, this enclosure is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. Accordingly, DEF respectfully requests that the information (Attachment A to Enclosure 1) which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Westinghouse's Application for Withholding Proprietary Information from Public Disclosure CAW-14-3961 and accompanying Affidavit, and Proprietary Information Notice and Copyright Notice are provided as Enclosure 2 and Enclosure 3 respectively.

Enclosure 4 contains the proposed changes to the AP1000 DCD Tier 2 licensing basis associated with the responses to RAI Questions 06.03-10, 06.03-11 and 06.03-12. Enclosure 5 contains the Levy Nuclear Plant Part 2 and Part 4 COL application (COLA) revisions based on the Enclosure 4 DCD changes, which will be included in a future update of the COLA.

If you have any further questions, or need additional information, please contact Bob Kitchen at (704) 382-4046, or me at (704) 382-9248.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on June 27, 2014

Sincerely,

Christopher M Fallon

Christopher M. Fallon
Vice President
Nuclear Development

Enclosures/Attachments:

1. Levy Nuclear Power Plant Units 1 and 2 Response to NRC Request for Additional Information Letters No. 116 Related to SRP Sections 06.03 and 15.02.06 for the Combined License Application, Dated March 6, 2014, No. 117 Related to SRP Section 06.03, Dated April 10, 2014, and No. 118 Related to SRP Section 06.03, dated April 24, 2014
   A. Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return Licensing Submittal (Proprietary)
   B. Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return Licensing Submittal (Nonproprietary)
2. Westinghouse Application Letter CAW-14-3961 and Affidavit
3. Proprietary Information Notice and Copyright Notice
4. AP1000 DCD Tier 2 Licensing Basis Document – Proposed Changes
5. Levy Nuclear Plant Units 1 and 2 - Part 2 and Part 4 COL Application Revisions

cc: U.S. NRC Region II, Regional Administrator
Mr. Donald Habib, U.S. NRC Project Manager

**Levy Nuclear Power Plant Units 1 and 2
Response to NRC Request for Additional Information Letters No. 116 Related to SRP Sections 15.02.06 and 06.03 for the Combined License Application, dated March 6, 2014, No. 117 Related to SRP Section 06.03, dated April 10, 2014, and No. 118 Related to SRP Section 06.03, dated April 24, 2014**

NRC RAI #	Duke Energy RAI #	Duke Energy Response
15.02.06-1	L-1081	NPD-NRC-2014-017, dated June 19, 2014
15.02.06-2	L-1082	Response enclosed – see following pages
15.02.06-3	L-1085	NPD-NRC-2014-017, dated June 19, 2014
06.03-1	L-1086	NPD-NRC-2014-014, dated May 5, 2014
06.03-2	L-1087	NPD-NRC-2014-016, dated June 12, 2014
06.03-3	L-1088	NPD-NRC-2014-016, dated June 12, 2014
06.03-4	L-1089	Future Response
06.03-5	L-1090	Response enclosed – see following pages
06.03-6	L-1091	NPD-NRC-2014-014, dated May 5, 2014
06.03-7	L-1092	NPD-NRC-2014-012, dated April 17, 2014
06.03-8	L-1093	NPD-NRC-2014-012, dated April 17, 2014
06.03-9	L-1094	NPD-NRC-2014-015, dated May 19, 2014
06.03-10	L-1096	Response enclosed – see following pages
06.03-11	L-1097	Response enclosed – see following pages
06.03-12	L-1099	Response enclosed – see following pages

NRC Letter No.: LNP-RAI-LTR-116

NRC Letter Date: March 6, 2014

NRC Review of Section 15.02.06 - Loss of Non-Emergency AC Power to the Station Auxiliaries

NRC RAI #: 15.02.06-2

Text of NRC RAI:

In DCD Section 6.3.1.1 it is stated that for postulated non-LOCA events, "The passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, is designed to remove decay heat for an indefinite time in a closed-loop mode of operation. The passive residual heat removal heat exchanger is designed to cool the reactor coolant system to 420 F in 36 hours, with or without reactor coolant pumps operating". DCD Section 6.3.4 states "The passive core cooling system can maintain safe shutdown conditions for 72 hours after an event without operator action and without both nonsafety-related onsite and offsite power." Previous approval of Chapter 15 analyses in the DCD were based on the assumption that the PRHR-HX would operate indefinitely. Therefore, the calculation could be terminated once the acceptance criteria for the design basis event were initially met. The current submittal has revealed that the assumption of indefinite operation in not valid. In order to provide safe closure of the Chapter 15 events, staff needs to verify that the acceptance criteria for these events will continue to be satisfied. The staff requests the following addition information:

a. Provide an explanation as to why loss of normal feedwater coincident with loss of ac power is the limiting event from the standpoint of PRHR-HX performance.
b. Extend the calculation time for the limiting event to 72 hours. In addition to the plots already presented in the DCD include plots for (1) IRWST level as a function of time, (2) condensate return fraction as a function of time, and (3) containment pressure as a function of time.

DEF RAI ID #: L-1082

DEF Response to NRC RAI:

See Attachment A for the proprietary version of the response to NRC RAIs 06.03-5 and 15.02.06-2.

See Attachment B for the nonproprietary, redacted version of the response to NRC RAIs 06.03-5 and 15.02.06-2.

Associated LNP COL Application Revisions:

None

Attachments/Enclosures to Response to NRC:

A. Proprietary version of response to RAIs 06.03-5 and 15.02.06-2

B. Nonproprietary version of response to RAIs 06.03-5 and 15.02.06-2

NRC Letter No.: LNP-RAI-LTR-116

NRC Letter Date: March 6, 2014

NRC Review of Section 06.03 - Emergency Core Cooling System

NRC RAI #: 06.03-5

Text of NRC RAI:

Condensate losses over attachments to the containment wall are based on testing reported in TR-SEEE-III-12-01. Tests were done over different attachment plate types at varying flow rates at both room temperature and heated conditions. Because conditions inside containment following a postulated transient analyzed in APP PXS M3C-072 ("Condensate Return to IRWST for Long Term PRHR Operation"), Revision 1, result in rather higher temperatures than those observed in the tests, the losses over attachments to the containment wall were extrapolated from the test results. Provide a justification for the validity of the extrapolation and explain the impact of the calculated condensate loss rates on the return of water to the IRWST.

DEF RAI ID #: L-1090

DEF Response to NRC RAI:

See Attachment A for the proprietary version of the response to NRC RAIs 06.03-5 and 15.02.06-2.

See Attachment B for the nonproprietary, redacted version of the response to NRC RAIs 06.03-5 and 15.02.06-2.

Associated LNP COL Application Revisions:

None

Attachments/Enclosures to Response to NRC:

A. Proprietary version of response to RAIs 06.03-5 and 15.02.06-2

B. Nonproprietary version of response to RAIs 06.03-5 and 15.02.06-2

NRC Letter No.: LNP-RAI-LTR-117

NRC Letter Date: April 10, 2014

NRC Review of Section 06.03 - Emergency Core Cooling System

NRC RAI #: 06.03-10
Text of NRC RAI:

As stated in section 4.3.3.5 of the Utility Requirements Document (URD) and restated in Section 2.3.2 of the staff's safety evaluation, a design requirement for the passive decay heat removal system is to have sufficient water capacity in the passive decay heat water pools to permit 72 hours of operation after SCRAM without the need for refill (ADAMS Accession No. ML070600372). Based upon the licensing guidance in the URD, NUREG-1242, SECY-94-084, and the Regulatory Treatment of Non-Safety Systems (RTNSS) as discussed in the Section 19.3 of the Standard Review Plan, in order for the Passive Residual Heat Removal Heat Exchanger (PRHR-HX) to meet the requirements of GDC 34 and GDC 44, the In-containment Refueling Water Storage Tank (IRWST) should have sufficient capacity to permit a minimum of 72 hours of operation after SCRAM following an accident without the need for refill. The submitted changes to the passive core cooling system regarding condensate return has caused staff to question the mission time for the PRHR-HX/IRWST. The staff requests the following:

- What is the safety-related mission time for the PRHR-HX/IRWST following a non-LOCA accident?
- Provide the PRHR-HX tube plugging assumption used in the analysis of design basis accidents in Chapter 15 that credit use of the PRHR-HX.
- Provide the PRHR-HX tube plugging assumption used in the safe-shutdown analysis presented in Appendix E of Chapter 19.
- Update the FSAR to clarify the safety-related design basis for the PRHR-HX/IRWST regarding the 72 hour capacity of the IRWST for the mitgation of accidents.

DEF RAI ID #: L-1096

DEF Response to NRC RAI:

PRHR HX Safety-Related Mission Time: The **AP1000** plant meets the requirements of the Commission's regulations as well as the EPRI URD design requirements. The URD requires advanced light water reactor designs provide for mitigation of transients and accidents to meet the requirements of GDCs 34 and 44 for at least 72 hours without the need for operator action to replenish the water pools providing decay heat sink function (the passive containment cooling water storage tank or the in-containment refueling water storage tank). The plant is equipped with diverse means of meeting this requirement – closed-loop cooling using the PRHR HX and open-loop cooling pairing automatic depressurization system (ADS) valve operation and IRWST injection.

- The preferred means of providing long-term residual heat removal following a non-LOCA event is closed-loop cooling using the PRHR HX. As described in DCD/UFSAR subsection 6.3.2.1.1, "Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions," the PRHR HX provides emergency core decay heat removal for events not involving a loss of coolant. Closed-loop PRHR HX cooling is the primary means of meeting the Condition I, II, III, and IV success criteria: preventing fuel rod failures, preventing reactor coolant system (RCS) failures, and preventing secondary system overpressurization. Section 7.4 of the DCD/UFSAR notes the "safe shutdown conditions" for the RCS are those in which the reactor is subcritical, stable, and borated; RCS average temperature is less than or equal to no-load average temperature; and adequate RCS inventory and core cooling are established. The PRHR HX operates to bring the RCS to and maintain the RCS in an acceptable, stable condition for at least 72 hours after a non-LOCA event to allow ample time for decision-making and initiation of recovery actions. Subsection 6.3.1.1.1 of the plant-specific DCD/UFSAR, in which the passive core cooling system safety-related emergency core decay heat removal design bases are enumerated, is changed to clarify the safety-related mission time for the PRHR HX. This clarification specifies that, during this 72 hour time period, the applicable Chapter 15 design basis safety evaluation criteria are met. Fulfilment of this design requirement will be demonstrated using conservative, design basis assumptions and conditions.

- PRHR HX Tube Plugging: A design change was implemented to reduce the allowable number of plugged tubes for the PRHR HX from that making up 8 percent of the heat transfer area to a number of tubes making up 5 percent of the heat transfer area. This reduction in the allowable tube plugging was implemented in the revised analyses supporting the shutdown temperature evaluation presented in Appendix E of Chapter 19 of the submittal. The tube plugging reduction is not implemented in the Chapter 15 analyses. The PRHR performance modelled in the existing Chapter 15 analyses assumes 8 percent of the tubes are plugged. This tube plugging assumption is more conservative because plugging a larger portion of the tubes reduces heat removal by the heat exchanger, which presents greater challenge to meeting the design basis event safety evaluation success criteria. However, note that for those design basis events in which high PRHR HX capacity is conservative (i.e., 0% tubes plugged is more conservative for the main steam line break event) the analysis assumes all of the PRHR HX tubes are available for heat transfer.

- 72 Hour Mitigation of Accidents: As described in the first part of this response, the PXS is automatically actuated to bring the plant to a safe shutdown condition after a design basis event. The PRHR HX is automatically actuated to provide core decay heat removal to meet event success criteria during design basis accidents; and with limited operator action to prolong its operation, can maintain acceptable RCS conditions for at least 72 hours. These are safety-related design requirements.

The duration the PXS can maintain a safe, stable condition is predicated on Passive Containment Cooling System (PCS) water storage tank capacity, which is verified and tested during preoperational testing as described in DCD/UFSAR subsection 6.2.2.4.2 and in accordance with surveillance requirements of subsection 3.6.6 of the Technical Specifications. No licensing basis changes are required to reflect this information.

The DCD/UFSAR sections describing the PRHR HX capabilities are updated to provide a statement concerning the safety-related mission time of the PRHR HX. A description of the plant cooldown by the safety-related systems to maintain a safe, stable condition for at least 72 hours after a design basis event without actuation of the automatic depressurization system was evaluated in the response to request for additional information (RAI) 7440, question 15.02.06-2 and will be discussed in the licensing basis. The duration the PRHR HX can maintain the Condition II success criteria is dependent on the efficiency of the condensate return features. The analysis assumes design basis inputs and assumptions. In addition, it will be noted in the licensing basis, that the event assumes operators have taken manual action to maintain the closed-loop cooling mode of operation beyond the automatic depressurization system actuation time. A description of the event and summary of the associated analysis will be included in DCD/UFSAR subsection 6.3.3.2.1 as a new subsection 6.3.3.2.1.1.

Changes to the AP1000 DCD Tier 2 licensing basis to clarify the safety-related mission time of the PRHR HX in the closed loop mode of operation are shown in Enclosure 4, while the associated LNP COL application revisions are shown in Enclosure 5.

Associated LNP COL Application Revisions:

See Enclosure 5

Attachments/Enclosures to Response to NRC:

See Enclosures 4 and 5

NRC Letter No.: LNP-RAI-LTR-117

NRC Letter Date: April 10, 2014

NRC Review of Section 06.03 - Emergency Core Cooling System

NRC RAI #: 06.03-11
Text of NRC RAI:

In letter NPD-NRC-2014-005, dated February 07, 2014, Section 1.0 of Enclosure 2 states that among the "safety-related" design bases of the Passive Core cooling System (PXS) is the capability of the Passive Residual Heat Removal Heat Exchanger (PRHR-HX) to cool the Reactor Coolant System (RCS) to the safe shutdown condition of 420 °F in 36 hours. Compliance with safety-related design requirements is typically demonstrated through the use of conservative analyses or best estimate plus uncertainty evaluations. The best estimate shutdown temperature evaluation provided in Section 19E.4.10.2 has caused staff to question whether the treatment of uncertainty is adequate to demonstrate the safety-related design basis of the PRHR-HX having the capability to cool the RCS to 420 °F in 36 hours. Staff requests the following:

- Provide the conservative assumptions used for the AP1000 Safe Shutdown Temperature Evaluation.
- If obtaining safe shutdown in 36 hours is a safety-related design requirement, update the FSAR with a conservative, design-basis analysis.
- If it is determined that obtaining safe shutdown in 36 hours is not a safety-related design requirement, provide justification and update the FSAR accordingly.

DEF RAI ID #: L-1097

DEF Response to NRC RAI:

The capability of the AP1000 plant to maintain adequate core cooling to meet the GDC 34 and 44 requirements in the long-term (for at least 72 hours) is demonstrated using bounding design basis assumptions as described in the response to RAI 7440, question 15.02.06-2. The capability to bring the plant to the specified long-term safe shutdown condition of 420°F within 36 hours after an event is not a Chapter 15 success criterion; but is still considered a safety-related design requirement. While only safety-related components are credited in the shutdown temperature evaluation, the capability to meet this design criterion is demonstrated using a conservative analysis that is not bounding in all aspects when compared to the Chapter 15 analyses. The conservatisms incorporated include not only conservative modelling assumptions; but conservatisms accounted for in development of the condensate return rate to the IRWST.

- The conservative assumptions used for the Shutdown Temperature Evaluation summarized in Chapter 19E include the following:
    - Condensate return rate

    - Testing uncertainty was accounted for in the development of the analytical condensate return fraction by applying a multiplier to the experimental losses such that the analytical losses bound the highest loss fractions observed by experiment.

- The LOFTRAN analysis of the loss of normal feedwater (LONF)/loss of ac power (LOAC) event incorporated the following conservatisms:
    - The steam generator mass at the time of reactor trip is modelled conservatively. Bounding steam generator level trips were credited for the low steam generator narrow range and wide range level trips.
        - Using the conservative low level reactor trip setpoint results in a steam generator mass at the time of reactor trip that is only 42 percent of the initial mass at the beginning of the transient.
    - Heat transfer from the RCS thick metal to the containment atmosphere is not credited.
        - This eliminates the effect of this heat removal mechanism, which maximizes the PRHR HX heat input and IRWST steaming.
    - Maximum initial fuel pellet average temperatures are modelled for the entire core.
        - Maximum core stored energy
    - Uncertainties on initial RCS pressure and temperature.
        - Nominal full power initial reactor vessel average coolant temperature plus uncertainty.
        - Nominal initial pressurizer pressure minus uncertainty was modelled to minimize the energy release from the RCS through the pressurizer safety valves.
        - Maximum initial pressurizer water volume at full power plus uncertainty.
    - PRHR HX and IRWST
        - Consistent with the Chapter 15 non-LOCA safety analysis, the vertical form of the Rohsenow heat transfer equation was used to determine the heat flux between the PRHR HX tube wall and the IRWST in nucleate boiling for the entire transient.
        - Maximum allowable number of PRHR HX tubes is assumed to be plugged.

    - The PRHR HX actuates on a conservatively low steam generator wide range level of approximately 13 percent of wide range span.

- The ambient air, which acts as the ultimate heat sink, is assumed to be a constant 115°F.
    - Natural temperature fluctuations of the ambient air reflecting night time and day time are not modelled.
- The initial containment temperature and the temperature of all the structures and components inside containment are assumed to be conservatively low when compared to the high ambient air temperature.
    - The lower containment temperature increases the amount of condensate formed on heat sinks, which does not return to the IRWST.

As previously mentioned, the capability to bring the plant to the specified long-term safe shutdown condition of 420°F within 36 hours after an event is not a Chapter 15 success criterion; but is still considered a safety-related design requirement. While only safety-related components are credited in the shutdown temperature evaluation, the capability to meet this design criterion is demonstrated using a conservative analysis that is not bounding in all aspects when compared to the Chapter 15 analyses. Namely, nominal decay heat and nominal full power levels are assumed for the Chapter 19E analysis. It is considered reasonable to demonstrate the capability for the PRHR HX to bring the plant to 420 in 36 hours using the thermal hydraulic analysis summarized in Chapter 19E, which is a non-bounding conservative analysis, for the following reasons:

- The PRHR HX is an extremely reliable, safety-related component.
    - The PRHR HX, its heat transfer capability and the materials of its construction are based on proven design.
    - The PRHR HX and the components supporting its operation are simple, single-failure resistant, and fail to their safe position when all power sources are lost.
    - The PRHR HX and the components supporting its operation are safety-related, designed to applicable nuclear safety codes and standards, and qualified to operate in a harsh environment.
    - The PRHR HX is automatically actuated. Two diverse systems can actuate the PRHR HX: the PMS and the diverse actuation system (DAS).
    - The PRHR HX can be initiated at full RCS design temperature and pressure. (RCS cooldown is not necessary for actuation of this feature.)
- The PRHR HX is backed up by a separate, diverse, safety-related residual heat removal system.

- The ADS actuation combined with passive safety injection can bring the plant to the specified safe shutdown condition.
- The ADS and passive safety injection use only safety-related components designed to applicable nuclear safety codes and standards, and qualified to operate in a harsh environment.
- ADS / passive safety injection are automatically actuated by the PMS, and can be manually actuated by the DAS.
- ADS / passive safety injection require only safety-related Class 1E dc power to assume their safe position.
- The ADS / passive safety injection can be initiated at full RCS design temperature and pressure. (RCS cooldown is not necessary for actuation of this feature.)

- The probability the PRHR HX would be unable to adequately perform the specified safe shutdown function, and open-loop cooling would be required to bring the plant to safe shutdown after a non-LOCA event is remote – on the order of 1E-07 per reactor year – much lower than the frequency of other initiating events that require ADS actuation, such as small break LOCA.
    - A loss of offsite power is the most frequent event that could challenge long-term PRHR HX operation. (The loss of normal feedwater with consequential loss of offsite power modelled in the shutdown temperature evaluation is less frequent.) A long-term loss of offsite power event would most likely be marked by the following:
        - Loss of offsite power (initiating event frequency of 1.20E-01 /year, DCD Rev. 19 PRA)
        - Failure of both diesel generators to operate (probability of 2.13E-3 /demand, DCD Rev. 19 PRA)
        - Failure to recover offsite power within 24 hours (probability of 1.79E−02 /demand, NUREG/CR-6890)
        - Decay heat as high as 2σ above nominal (probability of 2.3E-02 /demand, ANS-5.1-1979)
    - In the unlikely event all of the conditions listed above occur, the PRHR HX would still successfully perform its safety mission, bringing the RCS to an acceptable, stable condition. As demonstrated in the response to RAI 7440, question 15.02.06-2 for this scenario, the RCS would be stable with a temperature higher than 420°F. In this condition, the health and safety of the plant workers and the public will not be at risk. However, for the purpose of this evaluation, it is assumed the ADS could be actuated earlier than would be necessary.

  - This evaluation shows the probability the operators would actuate ADS to provide long-term core cooling due to PRHR HX performance that does not trend as expected is significantly smaller than for other sequences during which ADS would automatically be actuated. For example, the initiating event frequency for a small break LOCA, for which ADS actuation occurs as part of short-term accident mitigation, is 5.00E-04 /year (DCD Rev. 19 PRA).
- Establishing an RCS temperature of 420°F is not a prerequisite for maintaining safe, stable RCS conditions.
  - Maintaining a stable, post-accident condition with an RCS temperature higher than 420°F would not result in exceeding any of the safety evaluation criteria evaluated in the bounding, conservative Chapter 15 analyses.
    - The core would still be cooled.
    - RCS pressure would still be a small fraction of its design pressure.
    - Peak cladding temperatures, departure from nucleate boiling, and pressurizer level would be maintained within acceptable limits of the evaluation criteria.
  - The capability to maintain hot standby conditions for at least 8 hours after an event is a URD design requirement. A hot standby condition has also been acknowledged by the staff as a safe condition.
  - The Chapter 15 analyses demonstrate the plant is safe using fully conservative thermal hydraulic analyses. The limiting loss of main (normal) feedwater with loss of ac power event was extended to 72 hours as described in the response to request for information 7440, question 15.02.06-2. This analysis demonstrates the Chapter 15 evaluation criteria are met with extended PRHR HX operation. As described in the response to RAI 06.03-10, a summary of this transient is added in a new subsection of the licensing basis.

As stated in Section 7.4 of the DCD/UFSAR, the "safe shutdown conditions" for the RCS are those in which the reactor is subcritical, stable, and borated; RCS average temperature is less than or equal to no-load average temperature; and adequate RCS inventory and core cooling are established. The capability to bring the plant to a safe shutdown condition after an event is a safety-related design requirement. As discussed in the response to question 06.03-10, the **AP1000** plant is equipped with diverse, safety-related means of meeting this requirement.

The protection and safety monitoring system (PMS) provides for automatic actuation of the PRHR HX and the ADS to bring the plant to a safe shutdown condition within 36 hours after a design basis event. The automatically actuated means of achieving a safe shutdown condition after an event uses closed-loop and open-loop cooling: PRHR HX operation followed by automatic depressurization and steam relief through the ADS valves combined with passive injection of the IRWST water inventory into the RCS. The open-loop cooling mode uses single failure-tolerant, safety-related components that are diverse from the closed-loop mode of cooling. The open-loop cooling mode can be initiated at full RCS design pressure and temperature, a significant design improvement as compared to existing plants. The open-loop mode of cooling is automatically actuated by the PMS after a design basis event without the need for operator action for the first 72 hours. This mode of cooling can bring the plant to and maintain the plant in a safe shutdown condition indefinitely.

The PRHR HX is designed to provide at least 72 hours of core decay heat removal to meet the Condition II event success criteria in conjunction with the passive containment cooling system and to bring the plant to a safe shutdown condition without reliance on the ADS (closed-loop operation). Likewise, the closed-loop cooling mode uses single failure-tolerant, safety-related components, can be initiated at full RCS design pressure and temperature and can bring the plant to an acceptable, stable condition after a non-LOCA event. The closed-loop cooling mode using the PRHR HX can be maintained for more than 72 hours after an event to maintain the non-LOCA event success criteria and meet the Commission's regulations.

Closed-loop cooling using the PRHR HX is the preferred cooling mode for several reasons. Prolonging PRHR HX operation plays an important role in minimizing the need for ADS actuation. Avoiding open-loop cooling when it is not needed minimizes the occupational dose associated with recovery after the event and maintains radiation doses as low as reasonably achievable. Additionally, actuation of the ADS bypasses one of the defense-in-depth barriers to radiation release. Therefore, the frequency of ADS actuation is limited to a low probability to reduce those safety risks and to minimize plant outages. Per DCD/UFSAR subsection 1.2.1.4.1, "Engineered Safeguards Systems Design," the probability of significant flooding of the containment due to ADS actuation is limited to once in 600 reactor-years. Prolonging the duration and capability to maintain PRHR HX operation minimizes potential for depressurization transients and the associated outage time.

However, maintaining this closed-loop mode of PRHR HX operation for 72 hours is not the automatic post-accident protective action that would be initiated by the PMS after a non-LOCA event. As explained in DCD/UFSAR subsection 15.0.13, the PRHR HX automatically establishes an acceptable, stable condition after a non-LOCA event, at which point the plant operator is expected to take manual control of the plant and proceed with an orderly shutdown. In the event a loss of ac power or other associated non-LOCA event continued beyond several hours, operator assessment and action would be required to prolong PRHR HX operation beyond the automatic ADS actuation time. The Class 1E dc batteries supplying motive power to the ADS valves supply power for at least 24 hours. Upon loss of ac power, the input voltage to the Class 1E battery chargers would be lost, which would initiate an ADS actuation countdown timer within the PMS to ensure ample Class 1E motive power is available for actuation of the ADS. (See the description of this timer function in DCD/UFSAR subsection 6.3.7.7, "Automatic Depressurization System Actuation at 24 Hours.") The post-event operator assessment and action is described in DCD/UFSAR subsection 7.4.1.1, "Safe Shutdown Using Safety-Related Systems" and in the response to RAI 7440, question 15.02.06-1, part b.

ADS actuation may become required during long-term PRHR HX operation to cope with RCS leakage or in the case of a rare, severe non-LOCA event (massive tornado, beyond safe shutdown earthquake) after which there is a prolonged delay in recovery of an ac power source. The emergency procedures are based on realistic analyses of the dynamic plant response; and direct operators to make an assessment of plant conditions to determine whether the conditions for preserving and extending closed-loop cooling have been met. The operators will not de-energize the 24-hour battery powered loads if the realistic plant response is not indicative of unambiguous progression toward a sustainable safe shutdown condition. Likewise, if the 24-hour battery loads have been de-energized in order to extend closed-loop PRHR HX operation, and plant conditions unexpectedly degrade, the operators will re-energize the 24 hour battery loads and the plant will transition to the open-loop cooling mode, which will maintain safe shutdown conditions indefinitely.

The design basis analyses incorporate bounding, conservative assumptions that demonstrate the plant meets the safety evaluation success criteria, even with a worst-case plant response. For example, the design basis conservative assumptions applied to the loss of normal feedwater analysis result in higher reactor coolant system temperatures and higher pressurizer levels than an operator is expected to see during the event. While some of the assumptions of the shutdown temperature evaluation do not reflect the Chapter 15 design basis analysis assumptions, the shutdown temperature evaluation is still a conservative description of the expected plant response. As demonstrated in the early part of this response, conservatisms were incorporated into the safe shutdown temperature evaluation to account for uncertainties. Extended closed-loop PRHR HX operation to bring the plant to 420°F within 36 hours is modelled in a non-bounding, conservative analysis in order to better demonstrate the expected plant, and operator, response.

Changes to the AP1000 DCD Tier 2 licensing basis to clarify the basis for the safe shutdown analysis presented above are shown in Enclosure 4, while the associated LNP COL application revisions are shown in Enclosure 5.

Associated LNP COL Application Revisions:

See Enclosure 5

Attachments/Enclosures to Response to NRC:

See Enclosures 4 and 5

NRC Letter No.: LNP-RAI-LTR-118

NRC Letter Date: April 24, 2014

NRC Review of Section 06.03 - Emergency Core Cooling System

NRC RAI #: 06.03-12
Text of NRC RAI:

10 CFR Part 50, Appendix A, General Design Criteria 34 requires a system be provided with the safety function to transfer decay heat from the reactor core. In the AP1000 DCD, the passive residual heat removal heat exchanger (PRHR HX) is credited with performing this function in Chapters 6, 15, and 19.

In DCD Section 6.3.1.1, it is stated that for postulated non-LOCA events, "The passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, is designed to remove decay heat for an indefinite time in a closed-loop mode of operation." The current submittal has raised questions about connotations associated with "indefinite operation" of the PRHR HX in this context.

DCD Section 6.3.4 states that "the passive core cooling system can maintain safe shutdown conditions for 72 hours after an event without operator action and without both nonsafety-related onsite and offsite power." Pursuant to staff guidance in Section 4.3.3.5 of the Utility Requirements Document and consistent with Regulatory Treatment of Non-Safety Systems (RTNSS) as discussed in the Section 19.3 of the Standard Review Plan, in order for the PRHR HX to meet the requirements of GDC 34, the system should have sufficient capacity to permit a minimum of 72 hours of operation without operator action following an accident. A preliminary analysis of the calculations available for staff audit indicate the system, with the proposed changes, appears to be capable of performing its safety function for substantially longer than 72 hours.

Staff seeks to clarify the intent of the phrase "indefinite operation" in the context of the proposed design change. Please provide, in an RAI response, a more detailed explanation on the intent and meaning of "indefinite operation" as it applies to the non-LOCA events.

DEF RAI ID #: L-1099

DEF Response to NRC RAI:

The PRHR HX is credited with transferring decay heat from the reactor core to satisfy the acceptance criteria for Condition I, II, III, and IV events. The passive containment cooling water storage tank (PCCWST) holds 72 hours worth of water to cool the containment vessel and transfer energy to the ultimate heat sink (the ambient atmosphere) after a design basis event. With simple operator action to maintain that water supply – either from the permanently installed ancillary equipment, or from an offsite, pumped water supply – the PRHR HX can supply adequate core cooling for a significant duration beyond 72 hours.

Closed-loop cooling capacity is described in the current licensing basis as indefinite. That determination, as well as the design basis analyses and the safe shutdown temperature evaluation contained in the DCD/UFSAR, were based on the constant condensate return fraction. The indefinite operation was described in terms of the apparent capacity of this equipment in relation to the duration of the design basis analysis runs. The limiting design basis event with respect to challenging long-term the PRHR HX operation is the loss of ac power to the plant auxiliaries event. As described in DCD/UFSAR subsection 15.2.6.1, the transient analysis of this event was run for 36,000 seconds (10 hours). Evaluation of PRHR HX capability on this time scale showed performance levels indicative of cooling capacity far beyond that analyzed, upward of 40 days or longer. The capacity of the closed-loop cooling operation was characterized as indefinite based on those results.

As previously described, with operation action to extend its operation, the PRHR HX can maintain acceptable RCS conditions for at least 72 hours after a design basis event. The duration the PRHR HX can maintain the specified safe shutdown condition was determined using non-bounding, conservative assumptions to account for uncertainties. This evaluation indicates the PRHR HX can maintain the RCS temperature below 420°F for significantly longer than 14 days following a non-LOCA event. Sustaining this condition is predicated on the engagement of post-72 hour equipment to maintain the passive containment cooling system water storage tank water inventory beyond that provided by the 72-hour supply of water already in the tank.

Describing the duration of closed loop as indefinite can lead to potential misunderstandings. As a result, the licensing basis will be clarified to incorporate a quantitative basis for PRHR HX capacity. A capacity of 14 days of closed-loop cooling operation is considered adequate based on the following insights and operating experience:

- Condensate return to the IRWST may not be the limiting factor in the capacity for closed-loop operation. This mode of operation could be limited due to RCS leakage.
- The probability of a station blackout lasting longer than 14 days is remote. On average, offsite power is restored in about 2.5 hours after a loss of offsite power event. The duration of loss of offsite power events experienced by nuclear power plants in the United States range from a few hours to 5 days.
- The events that might result in extended loss of ac power are associated with severe, and likely beyond design basis, events, such as massive earthquakes or tornadoes. As an example, after Hurricane Andrew, offsite power was not restored to Turkey Point Units 3 and 4 for more than 5 days. At Fukushima Daiichi, partial offsite power was restored in 12 days.

The capability to provide at least 14 days of closed-loop cooling provides a sufficiently long period to allow recovery of ac power and the defense-in-depth equipment needed to transition to cold shutdown after all but the rarest and most severe non-LOCA events. In the event ac power cannot be restored in this time, the plant can still safely transition to open-loop cooling.

The capability to maintain closed-loop PRHR HX cooling for a period of greater than or equal to 14 days would exceed the regulatory requirements and expectations for long-term, closed-loop cooling. The **AP1000** plant is required to perform its safety functions for 72 hours after a design basis event without reliance on non-safety related systems, structures or components (SSCs) for support. The PRHR HX is adequate to perform this safety mission. Additionally, open-loop cooling is available to maintain acceptable RCS conditions indefinitely and independently of closed-loop cooling via the PRHR HX. Therefore the capability for the PRHR HX to maintain safe shutdown conditions for more than 14 days in a closed-loop mode of operation is considered a non-safety related design goal. This capability maintains the design goals for the probability for significant containment floodup (see the response to 06.03-11) and serves as investment protection by minimizing the probability of significant post-event outage time.

Although the capacity of the closed-loop mode of operation is not considered indefinite, the capability of the PXS to maintain safe shutdown conditions indefinitely is preserved because, as explained in the response to RAI 06.03-11, the open-loop cooling mode of operation can be manually actuated at any time by the operators and is a safety-related means of providing core cooling.

Changes to the AP1000 DCD Tier 2 licensing basis to clarify the intent of the phrase "indefinite operation" and define the duration of closed-loop mode of operation are shown in Enclosure 4, while the associated LNP COL application revisions are shown in Enclosure 5.

Associated LNP COL Application Revisions:

See Enclosure 5

Attachments/Enclosures to Response to NRC:

See Enclosures 4 and 5

Serial: NPD-NRC-2014-021

Attachment B to Enclosure 1 (Nonproprietary)

(42 pages including cover page)

Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return

June 2014

Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return

Information Requested:

06.03-5

Condensate losses over attachments to the containment wall are based on testing reported in TR-SEEE-III-12-01. Tests were done over different attachment plate types at varying flow rates at both room temperature and heated conditions. Because conditions inside containment following a postulated transient analyzed in APP-PXS-M3C-072 ("Condensate Return to IRWST for Long Term PRHR Operation"), Revision 1, result in rather higher temperatures than those observed in the tests, the losses over attachments to the containment wall were extrapolated from the test results. Provide a justification for the validity of the extrapolation and explain the impact of the calculated condensate loss rates on the return of water to the IRWST.

Response Information:

References

1) Hartley, D. and Murgatroyd, W., "Criteria For The Break-Up Of Thin Liquid Layers Flowing Isothermally Over Solid Surfaces", Int. J. Heat Mass Transfer, Vol. 7, 1964
2) Bird, R.; Stewart, W.; and Lightfoot, E., "Transport Phenomena", Wiley, 1960
3) Azbel, D. and Cheremisinoff, N., "Fluid Mechanics and Unit Operations", Ann Arbor Science, 1983
4) Doniec, A., "Flow Of A Laminar Liquid Film Down A Vertical Surface", Chemical Engineering Science, Vol. 43, No. 4, 1988
5) Simon, F. and Hsu, Y., "Thermocapillary Induced Breakdown Of A Falling Liquid Film", NASA Technical Note, NASA TN D-5624, 1970
6) Haocui Zhanga ,Jun Yue , Guangwe Chen, Quan Yuan "Flow pattern and break-up of liquid fi lm in single-channel falling film microreactors" Chemical Engineering Journal 163 (2010) 126-132
7) John R. Taylor, "An Introduction to Error Analysis" Second Edition
8) APP-PXS-M3C-072, Revision 1, "Condensate Return to IRWST for Long Term PRHR Operation"
9) Incropera F., DeWitt D "Fundamentals of Heat and Mass Transfer" Fifth Edition

In the following, the first section of this RAI response will explain the extrapolation of the condensate return attachment plate loss. The second section shows how experimental data base for the condensate loss has been used for the design basis and best estimate analysis. The third section will show the sensitivities associated with the condensate return attachment plate loss.

1) <u>Extrapolation of Condensate Return Attachment Plate Loss Testing to the **AP1000** Plant</u>

Condensate losses from attachment plates on the containment vessel wall are dependent upon the following key scaling and geometry-related parameters:

- Geometry (i.e. attachment plate shape and thickness, corners) and orientation (angle of inclination) of the containment vessel wall and attachment plate.
- Surface characteristics (i.e. wetting angle) of wall and plate surfaces.

[

]a,b,c

Conservative application of these key scaling and geometry-related parameters are important to support application of the containment vessel wall condensate return tests to the **AP1000** plant.

[

]a,b,c

a,b,c

Figure 1: Condensate Loss vs. Film Reynolds Number

Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return

Figure 1 shows that the condensate return test facility covered a range of Reynolds numbers that bounds the expected range of the **AP1000** plant. [

]a,b,c

The film Reynolds number for film flow is defined below and can be obtained from Nusselt's theory for film condensation (references 2, 3):

$$Re = \frac{4\Gamma}{\mu} \tag{1}$$

Where:

$\Gamma = mass\ flow\ rate\ of\ film\ per\ unit\ width$ [lbm/(ft*s)]

$\mu = viscosity\ of\ liquid\ film$ [lbm/(ft*s)]

[

]a,b,c

The transition from film regime to rivulet has been studied by several investigators (references 1, 4, 5) for falling film flow down vertical and inclined plates with various surface characteristics. Some correlations for film to rivulet transition have been developed to address the impact of surface characteristics via inclusion of wetting angle (based upon static contact lines between liquid, solid, and gas interfaces), however, this has proved challenging to model especially for heterogeneous (i.e. non-uniform) surfaces and dynamic contact lines where film breakup occurs. [

]a,b,c

a,b,c

[ ] a,b,c

a,b,c

Figure 2 shows the comparison of non-dimensional film thickness versus film Reynolds number for the condensate return test facility and the **AP1000** design conditions. The figure shows that at the same film Reynolds number, non-dimensional film thickness is greater in the condensate return test facility than for the **AP1000** design conditions. Hence, % losses at the attachment plates in the film regime are conservatively larger in the condensate return test facility.

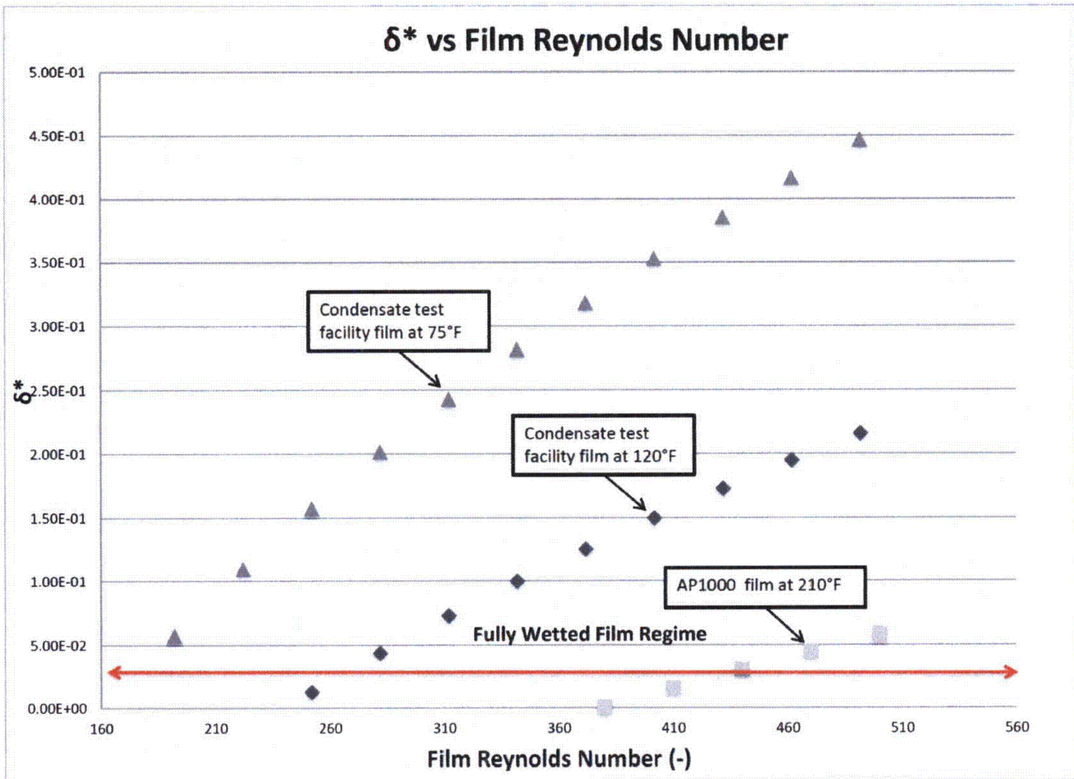

Figure 2: Non-Dimensional Film Thickness Comparison

In summary, justifying extrapolation of test results in Reference 8 for temperature extrapolation to the **AP1000** design is justified in that the geometry and surface related characteristics are preserved and the film Reynolds number bounded in the condensate return test, the attachment plate condensate loss data is applicable to the **AP1000** design. Less than prototypic condensate temperature in the test facility causes larger film thickness at a given film Reynolds number and leads to transition from film to rivulet regime at lower Reynolds numbers in the test relative to the **AP1000** plant. From scaling non-dimensional film thickness, the attachment plate detachment percent and loss data obtained in the condensate return test facility for the film regime is conservatively higher than that expected for the **AP1000** plant.

2) Attachment plate losses used for the design basis (DBA) and best estimate (BE) analysis

The previous section demonstrates that the percent loss data obtained in the condensate return test facility for the film regime is conservative. In addition, as shown below, additional conservatism has been added for the purpose of the analysis. The following two tables explain how the test data has been used for the purpose of the analysis and where the additional conservative factor has been added.

Table 1 shows the test data with the associated uncertainty for the cold and heated tests:

a,b,c

Considering the section 1 conclusion that the test results are conservative, the losses listed in the Table 1 for 2 gpm could have been directly used from the experimental database, without additional margins.

3) Sensitivity Results

Prior to analyzing the results of the sensitivity analysis, it is important to notice that the total attachment plate losses used for the design purposes are due to the:

III. Losses on the attachment plate
IV. Losses on the structure attached to the attachment plate

The Figure 3 illustrates an attachment plate and the structure attached to the attachment plate:

Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return

a,b,c

a,b,c

a,b,c

Table 5: Best estimate sensitivity results

For the previous runs, reactor vessel steaming was credited, initial heat sinks and initial IRWST temperature, consistent with the best estimate analysis, was set up at 85°F. For the best estimate case, time to uncover the PRHR is about 16.7 hrs and time to reheat the reactor coolant system (RCS) is about 20.4 days. For the most penalizing case, Sensitivity 3, time to uncover the PRHR tubes dropped to 16.2 hrs, which is about 30 min earlier. The time to reheat the RCS dropped from 20.4 days down to 18.8 days, which is less than 2 days. Though Sensitivity 3 reflects unrealistic attachment plate losses, even in this case, the impact on the overall results is small. Though Table 5 shows only the results for the BE case, the same delta in the results is expected for the DBA case.

Conclusion:

The margin added to the test results was larger than it needed to be, especially at 2 gpm. In fact, test results for the condensate losses, with the associated uncertainty, for the 2 gpm case were adequate for the DBA and BE analyses without the additional factor that was added. The margin added to the already conservative test results provides an additional conservatism and as a result, the condensate losses on the containment wall used for both DBA and BE analysis are adequate and justified.

Although the results were conservative, sensitivity studies were run with even higher losses. As shown in the 3 sensitivity studies, even with higher losses the duration of PRHR HX operation would only decrease slightly and would still meet the acceptance criteria.

Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return

Information Requested:

15.02.06-2

In DCD Section 6.3.1.1 it is stated that for postulated non-LOCA events, "The passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, is designed to remove decay heat for an indefinite time in a closed-loop mode of operation. The passive residual heat removal heat exchanger is designed to cool the reactor coolant system to 420 F in 36 hours, with or without reactor coolant pumps operating". DCD Section 6.3.4 states "The passive core cooling system can maintain safe shutdown conditions for 72 hours after an event without operator action and without both nonsafety-related onsite and offsite power." Previous approval of Chapter 15 analyses in the DCD were based on the assumption that the PRHR-HX would operate indefinitely. Therefore, the calculation could be terminated once the acceptance criteria for the design basis event were initially met. The current submittal has revealed that the assumption of indefinite operation in not valid. In order to provide safe closure of the Chapter 15 events, staff needs to verify that the acceptance criteria for these events will continue to be satisfied. The staff requests the following addition information:

a. *Provide an explanation as to why loss of normal feedwater coincident with loss of ac power is the limiting event from the standpoint of PRHR-HX performance.*
b. *Extend the calculation time for the limiting event to 72 hours. In addition to the plots already presented in the DCD include plots for (1) IRWST level as a function of time, (2) condensate return fraction as a function of time, and (3) containment pressure as a function of time.*

Response Information:

a. The following details why the loss of normal feedwater (LONF) with coincident loss of ac power (LOAC) is the limiting event from the standpoint of passive residual heat removal heat exchanger PRHR-HX performance.

Evaluation of Chapter 15 Non-LOCA Events

The following categories as detailed in the Design Control Document (DCD)/Updated Final Safety Analysis Report (UFSAR) are used to evaluate the events:

<u>**Increase in Heat Removal From the Primary System (15.1)**</u>

Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature (15.1.1)

This event is less severe than the increase in feedwater flow event or the increase in secondary steam flow event. [

]^{a,c} This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Feedwater System Malfunctions that Result in an Increase in Feedwater Flow (15.1.2)

The results of this event show that departure from nucleate boiling (DNB) did not occur at any time during the transient. As documented in the DCD/Updated Final Safety Analysis Report (UFSAR) for this event, following the reactor trip the plant approaches a stabilized and safe condition. [

]^{a,c} This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return

Excessive Increase in Secondary Steam Flow (15.1.3)

This event is analyzed to demonstrate that the new equilibrium conditions established from the increased secondary steam flow does not result in DNB. As no reactor trip is typically credited and the new equilibrium reaches a stabilized condition, this event does not degrade the secondary system from removing heat. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Inadvertent Opening of a Steam Generator Relief or Safety Valve (15.1.4)

This event is analyzed to demonstrate that the DNB safety analysis limit value is met by showing that no significant return to power occurs following the inadvertent opening of a steam generator relief or safety valve. [

,]^{a,c} this event is bounded by the "Steam System Piping Failure" event, which has a faster reduction in steam generator inventory and has a more limiting pressurization of containment.

Steam System Piping Failure (15.1.5)

The steam line break event is analyzed for DNB and the radiological consequences. The steam line break is currently categorized as an "Increase in Heat Removal From the Primary System" due to the initial cooldown caused by the break; however in the long term this event would decrease the heat removal from the primary system since the steam generators lose a significant amount of their inventory out the break. This event could challenge passive residual heat removal heat exchanger (PRHR-HX) performance due to the pressurization of containment which would result in a decrease in PRHR-HX performance. Thus a detailed analysis was performed in Appendix A. The Appendix A analysis demonstrates this event is less limiting than the "Loss of Normal Feedwater Without Offsite Power" event.

Inadvertent Operation of the PRHR Heat Exchanger (15.1.6)

This event is analyzed to demonstrate there is no violation of the core thermal design limits or RCS overpressure. This event generates an almost immediate reactor trip and does not degrade the secondary system from removing heat; thus, it is bounded by the "Decrease in Secondary System Heat Removal" category.

Decrease in Heat Removal by Secondary System (15.2)

Loss of External Electrical Load (15.2.2)

This event is analyzed for DNB and system pressure limits. As described in the DCD/UFSAR the loss of external load event is bounded by the turbine trip event as the primary pressure, temperature, and water volume transients are less severe due to a slightly slower loss of heat transfer capability. Additionally, the steam generators still have inventory available to provide cooling to the primary system.

Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return

Turbine Trip (15.2.3)

This event is analyzed for DNB and system pressure limits. The results of the analysis demonstrate that the DNB and pressure limits are met. This event is analyzed assuming no feedwater flow, startup feedwater flow or actuation of the PRHR-HX. Reactor trip occurs early at around 10 seconds and the transient is effectively over before initiation of the startup feedwater or the PRHR-HX. Since reactor trip occurs much earlier, and the steam generators still contain inventory for removing heat from the primary side, this event is less severe than the LOAC and LONF events in terms of the long term cooldown.

Inadvertent Closure of Main Steam Isolation Valves (15.2.4)

This event is bounded by the analysis of the Turbine Trip event.

Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip (15.2.5)

These events are bounded by the analysis of the Turbine Trip event.

Loss of ac Power to the Plant Auxiliaries (Loss of Normal Feedwater Flow without Offsite Power) (15.2.6)

This is the limiting event with respect to long term decay heat removal via the PRHR-HX. [

]a,c Thus, this event is the most limiting event with respect to the PRHR-HX performance.

[

]a,c

Loss of Normal Feedwater Flow (15.2.7)

Similar to the Loss of AC Power event, this event is analyzed to show that the PRHR-HX is capable of removing the stored and decay heat to prevent either overpressurization of the RCS or loss of water from the RCS via the pressurizer safety valves. However, for this event offsite power is assumed to be available and the reactor coolant pumps continue to run. [

]a,c This event is therefore less limiting than the "Loss of Normal Feedwater Without Offsite Power" event.

Feedwater System Pipe Break (15.2.8)

This event is analyzed to show the PRHR-HX is adequate to remove decay heat, to prevent overpressurizing the RCS, to maintain core cooling capability, and to limit the radioactivity doses to within the applicable limits. This event challenges PRHR-HX performance due to the fact the steam generators are significantly degraded. Thus, the PRHR-HX is the main source for long term cooling of the RCS. The analysis detailed in Appendix A demonstrates this event is less limiting than the "Loss of Normal Feedwater Without Offsite Power" event.

Decrease in Reactor Coolant System Flow Rate (15.3)

Partial Loss of Forced Reactor Coolant Flow (15.3.1)

This event is analyzed to show that the DNBR does not decrease below the safety analysis limit value. There is no degradation of the secondary system in removing heat from the primary system for this event. Thus both steam generators have full inventory to be used for the cooldown of the primary side after the trip. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Complete Loss of Forced Reactor Coolant Flow (15.3.2)

This event is analyzed to show that the DNBR does not decrease below the safety analysis limit value. There is no degradation of the secondary system in removing heat from the primary system for this event. Thus both steam generators have full inventory to be used for the cooldown of the primary side after the trip. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Reactor Coolant Pump Shaft Seizure (Locked Rotor) (15.3.3)

This event is analyzed to demonstrate that the peak RCS pressure, peak average clad temperature, and radiological dose limits are met. The locked rotor event assumes one RCP locks and the other RCPs continue to operate until the loss of ac power (LOAC) occurs from the turbine trip. Thus flow is rapidly reduced in the cold leg of one loop. [

][a,c] Therefore, this event is bounded by the "Decrease in Secondary System Heat Removal" category.

Reactor Coolant Pump Shaft Break (15.3.4)

This event is bounded by the Reactor Coolant Pump Shaft Seizure event.

Reactivity and Power Distribution Anomalies (15.4)

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition (15.4.1)

This event is analyzed to show that the DNBR does not decrease below the safety analysis limit value. As noted in the DCD/UFSAR, there is a high degree of subcooling at all times in the core. There is no degradation of the secondary system in removing heat from the primary system for this event. Thus both steam generators have full inventory to be used for the cooldown of the primary side after the trip. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.4.2)

This event is analyzed to show that the DNBR does not decrease below the safety analysis limit value. There is no degradation of the secondary system in removing heat from the primary system for this event. Thus both steam generators have full inventory to be used for the cooldown of the primary side after the trip. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Dropped RCCAs, Dropped RCCA Bank, and Statically Misaligned RCCA (15.4.3)

These events are analyzed to show that the DNBR does not decrease below the safety analysis limit value. There is no degradation of the secondary system in removing heat from the primary system for this event. After the rod drop, the plant will come to a new equilibrium condition or, for larger dropped rod worths, trip. Both steam generators have full inventory to be used for the cooldown of the primary side in the long term. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Single Rod Cluster Control Assembly Withdrawal (15.4.3.2.2)

This event is analyzed to show that the number of fuel rods experiencing DNB is less than 5 percent of the fuel rods in the core. There is no degradation of the secondary system in removing heat from the primary system for this event. Thus both steam generators have full inventory to be used for the cooldown of the primary side after the trip. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (15.4.4)

This event is prevented by administrative procedures.

A Malfunction or Failure of the Flow Controller in a Boiling Water Reactor Loop that Results in an Increased Reactor Coolant Flow Rate (15.4.5)

This is not applicable to the **AP1000** plant.

Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (15.4.6)

For cases initiated during startup or full power operation this event is analyzed to show that a post-trip return to criticality is prevented. There is no degradation of the secondary system in removing heat from the primary system for this event. Thus both steam generators have full inventory to be used for the cooldown of the primary side after the trip. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (15.4.7)

This event is prevented by administrative procedures.

Spectrum of Rod Cluster Control Assembly Ejection Accidents (15.4.8)

The rod ejection event is caused by a rupture of the control rod drive mechanism housing which results in a loss of coolant. As a result, operation of the PRHR HX will be limited by the actuation of automatic depressurization system (ADS). Therefore this event is not evaluated for long term PRHR-HX operation.

Increase in Reactor Coolant Inventory (15.5)

Inadvertent Operation of the Core Makeup Tanks During Power Operation (15.5.1)

This event is analyzed to show that the event does not propagate to a more serious event. There is no degradation of the secondary system in removing heat from the primary system for this event. Thus both steam generators have full inventory to be used for the cooldown of the primary side after the trip. In addition, even with the inadvertent actuation of the CMTs the passive safety systems (CMTs and PRHR-HX) continue to operate as designed to further cool the RCS. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory (15.5.2)

This event is analyzed to show that the event does not propagate to a more serious event. There is no degradation of the secondary system in removing heat from the primary system for this event. Thus both steam generators have full inventory to be used for the cooldown of the primary side after the trip. In addition, even with the actuation of the chemical and volume control system (CVS) the passive safety systems (CMTs and PRHR-HX) continue to operate as designed to further cool the RCS. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Boiling Water Reactor Transients (15.5.3)

This is not applicable to the **AP1000** plant.

Decrease in Reactor Coolant Inventory (15.6)

Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS (15.6.1)

The shorter term consequences of this event are analyzed to show that the DNBR does not decrease below the safety analysis limit value. There is no degradation of the secondary system in removing heat from the primary system for this event. Thus both steam generators have full inventory to be used for the cooldown of the primary side after the trip. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category.

Note: if in the long term the pressurizer safety valve or ADS valve are stuck-open, this event results in a loss of coolant. As a result, operation of the PRHR-HX will be limited by the actuation of the ADS. Therefore this event is not evaluated for long term PRHR-HX cooling.

Failure of Small Lines Carrying Primary Coolant Outside Containment (15.6.2)

This event is analyzed for radiological consequences. There is no reactor trip for this event. As such, the PRHR-HX is not actuated. This event is therefore bounded by the "Decrease in Secondary System Heat Removal" category for long term PRHR-HX cooling.

Steam Generator Tube Rupture (SGTR) (15.6.3)

This event is analyzed for steam generator overfill and radiological consequences. The results of the analysis demonstrate that steam generator overfill does not occur and doses are within the acceptance criteria. This event starts with a ruptured steam generator tube and an assumed LOAC. This results in a reactor trip. The main feedwater pumps coastdown following the LOAC. The steam generator break flow removes some energy from the primary side. However, in the long term the energy will be transferred back to the primary side and removed by the PRHR-HX. As discussed in the DCD/UFSAR the break flow is terminated without ADS actuation for this event. [

][a,c] As a result, this event is less severe than the LONF with coincident LOAC in terms of the long term PRHR-HX cooling.

Spectrum of Boiling Water Reactor Steam System Piping Failures Outside of Containment (15.6.4)

This is not applicable to the **AP1000** plant.

Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary (15.6.5)

This event results in a loss of coolant. As a result, operation of the PRHR-HX will be limited by the actuation of the automatic depressurization system (ADS). Therefore this event is not evaluated for long term PRHR-HX operation.

b. The most limiting event with respect to long term PRHR-HX performance is the loss of normal feedwater with loss of ac power (loss of ac power to the station auxiliaries) following reactor trip. This event was analyzed with Chapter 15 assumptions extended out to 72 hours. This analysis was run to demonstrate that the Chapter 15 acceptance criteria for the event (as described in DCD/FSAR Section 15.2.6) have been met and by doing so demonstrate the RCS is in an acceptable, stable condition. Achieving an RCS temperature of 420°F is not an acceptance criterion for the event. Below are the details of the loss of normal feedwater with coincident loss of ac power extended out to 72 hours.

The transient response of the reactor coolant system following a loss of normal feedwater with coincident loss of ac power is shown in Figures 1 through 15 including the in-containment refueling water storage tank (IRWST) level as a function of time (Figure 13), the condensate return fraction as a function of time (Figure 14), and the containment pressure as a function of time (Figure 15). The calculated sequence of events for this event is listed in Table 1. Table 2 presents the results for the acceptance criteria of interest.

The LOFTRAN code results show that the natural circulation flow and the PRHR system are sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown.

Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return

Immediately following the reactor trip on low narrow range span in the steam generator, the heat transfer capability of the PRHR-HX and the steam generator heat extraction rate are sufficient to slowly cool down the plant. The cooldown continues until a low T_{cold} setpoint is reached. As described in subsection 7.3.1.1 of the DCD/UFSAR, the low T_{cold} setpoint results in an "S" signal. The "S" signal actuates the core makeup tanks (CMTs). During this phase of the transient, actuation of the CMTs accelerates the cooldown of the plant. The CMT flow slowly decreases as the CMT fluid temperature increases due to water recirculation.

As a result of the cooldown by the CMTs, the heat removal by the PRHR-HX is lowered. As the temperature of the CMTs increases, the CMT circulation flow rate decreases and the CMT outlet temperature increases. At this time the total heat removal rate from the RCS decreases below the core decay heat produced and the RCS begins heating up again. As the RCS temperature increases, the heat removal by the PRHR-HX also increases. The RCS temperature slowly increases until the heat removal rate of the PRHR-HX matches the core decay heat produced, even as the IRWST water level slowly decreases due to the steam release from the IRWST and the amount of condensate returned back to the IRWST. As the horizontal tubes of the PRHR-HX uncover, the heat transfer capability is reduced. The RCS temperature slowly increases until it reaches a temperature where the heat removal rate of the PRHR-HX again matches the core decay heat produced. This cooldown continues until the end of the transient evaluation at 72 hours. The heat transfer capability of the PRHR-HX is sufficient to avoid event propagation. In conclusion, even with the upper horizontal section and a portion of the vertical tubes uncovered there is continued cooldown until the end of the transient and all Chapter 15 acceptance criteria of interest are met.

Table 1: Time Sequence of Events for LONF-LOAC Event

Sequence of Events	Time (seconds)
Feedwater is Lost	10.0
Low Steam Generator Water Level (Narrow-Range) Reactor Trip Setpoint Reached	58.0
Rods Begin to Drop	60.0
RCP trip due to loss of ac power	68.0
Low Steam Generator Water Level (Wide-Range) Reached	211.7
PRHR HX Actuation on Low Steam Generator Water Level	228.7
IRWST Reaches Saturation Temperature	~6200
Low T_{cold} Setpoint Reached	8624
Steam Line Isolation on Low T_{cold} Signal	8636
CMTs Actuated on Low T_{cold} Signal	8641
Maximum Pressurizer Water Volume Reached	38614
CMTs Stop Recirculating	40490

Table 2: Pertinent Results

Criterion	Value	Time (s)
Minimum DNBR Safety Analysis Limit = 1.50	2.028	61.0
Maximum Pressurizer Water Volume SAL = 2199.72 ft^3 (62.3 m3)	2041 ft^3 (57.79 m^3)	38614.0
Maximum RCS Pressure SAL = 2748.5 psia (189.50 bar)	2593 psia (178.7 bar)	28008.0
Maximum SG Pressure SAL = 1318.5 psia (90.91 bar)	1267 psia (87.33 bar)	69.0

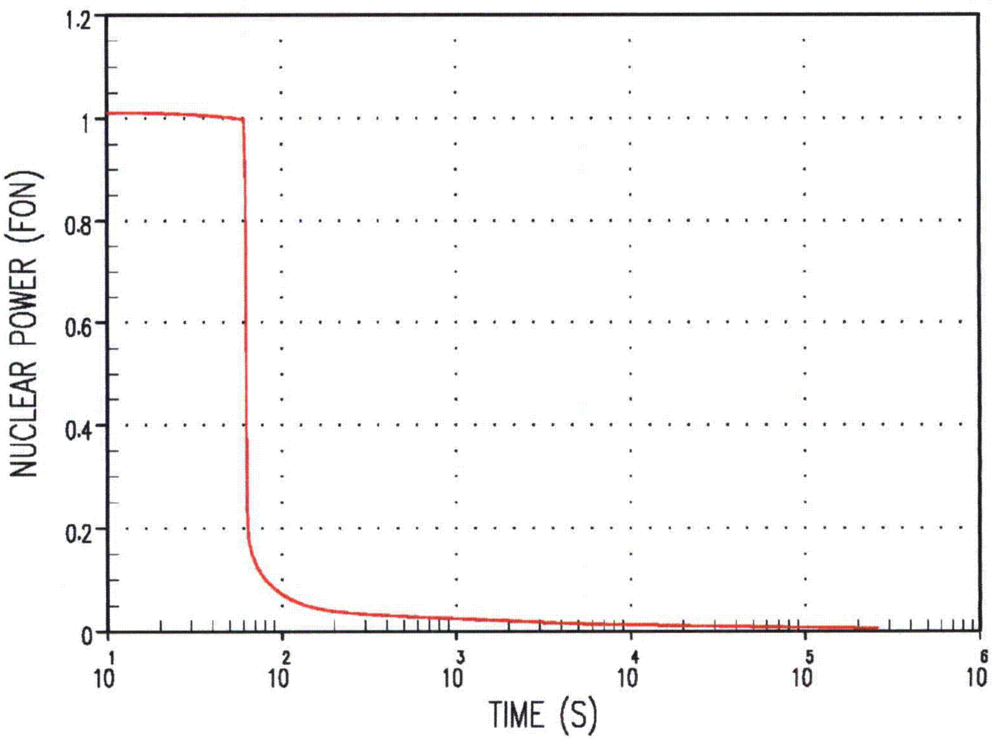

Figure 1: Nuclear Power

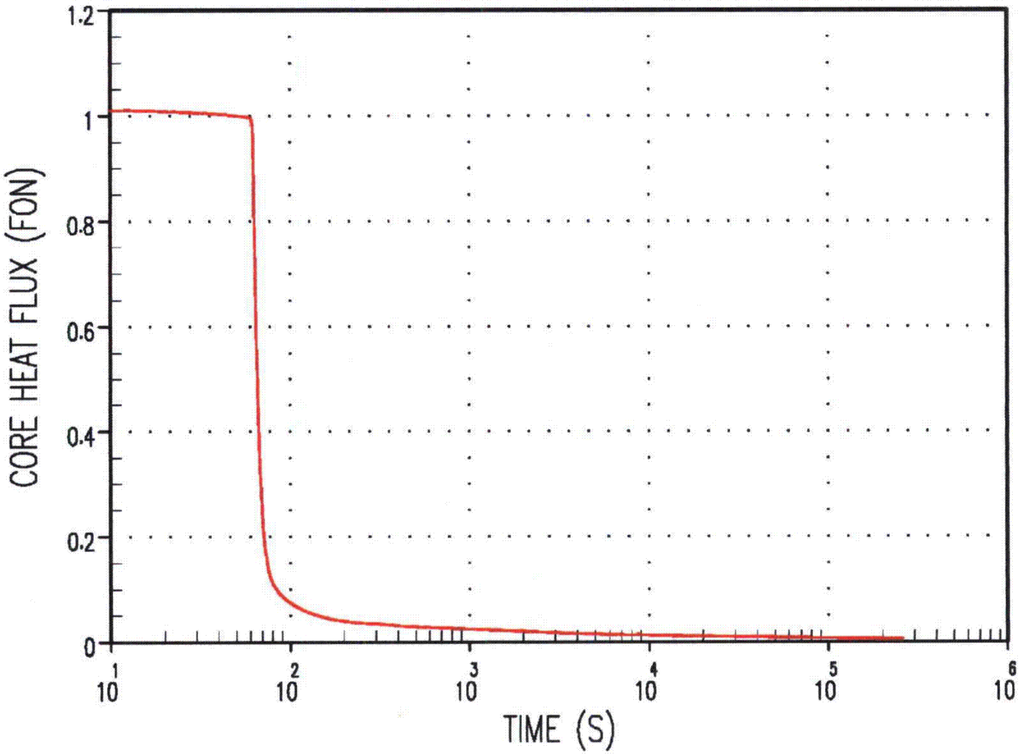

Figure 2: Core Heat Flux

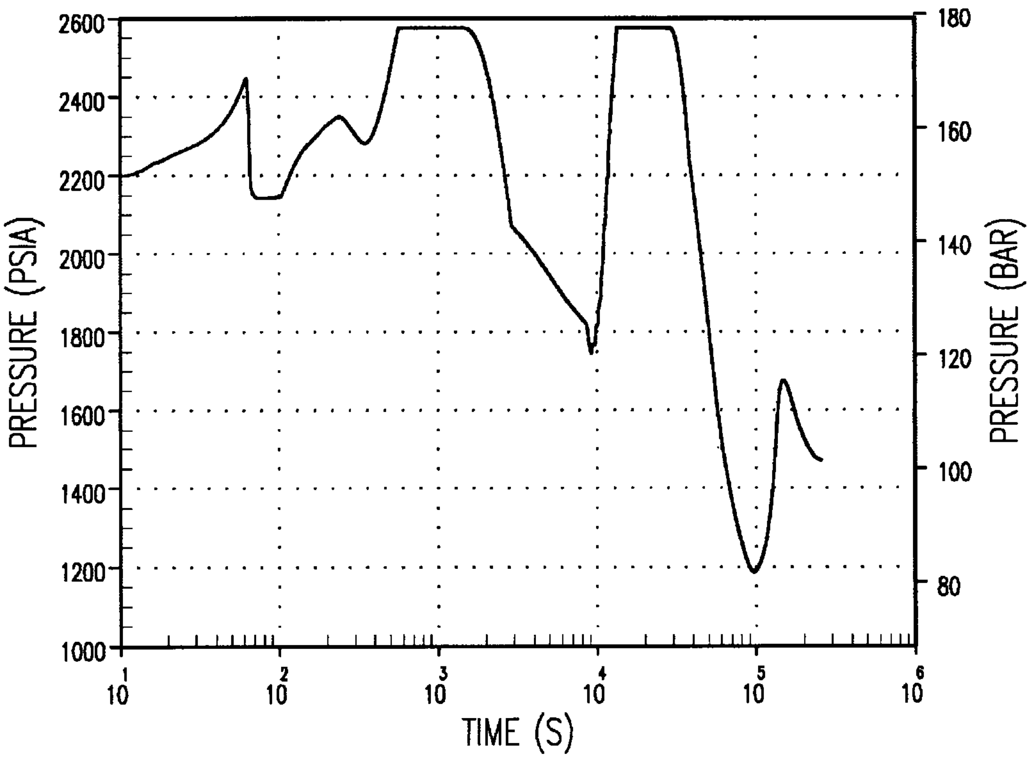

Figure 3: Pressurizer Pressure

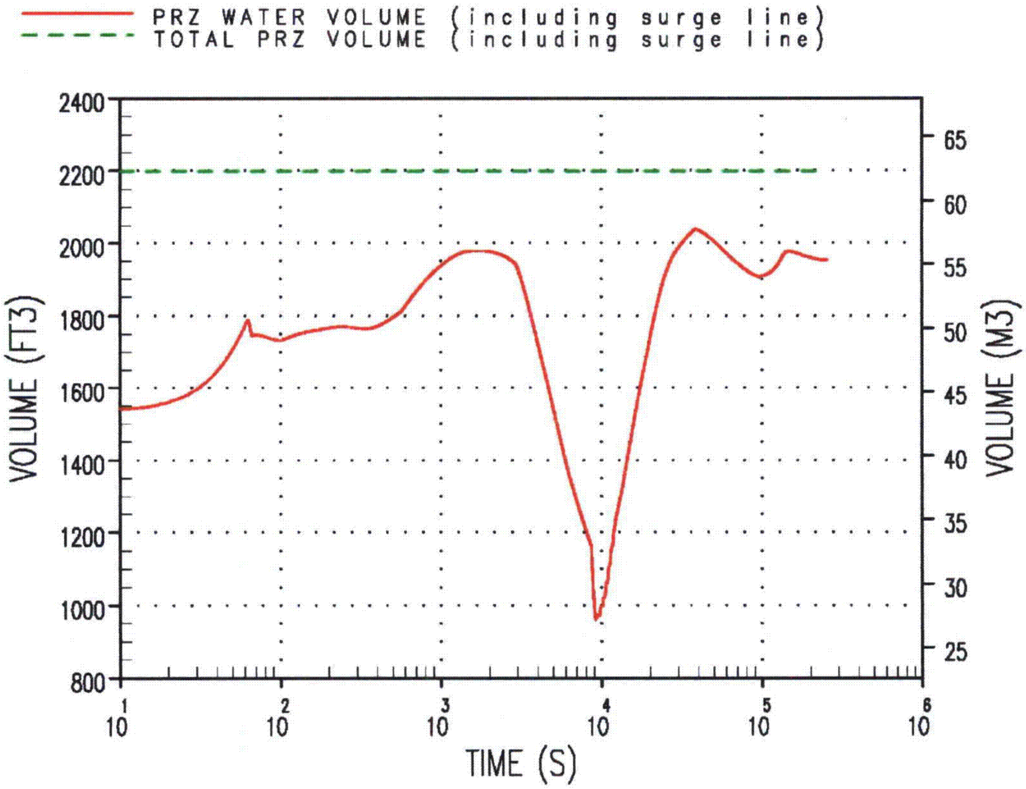

Figure 4: Pressurizer Water Volume

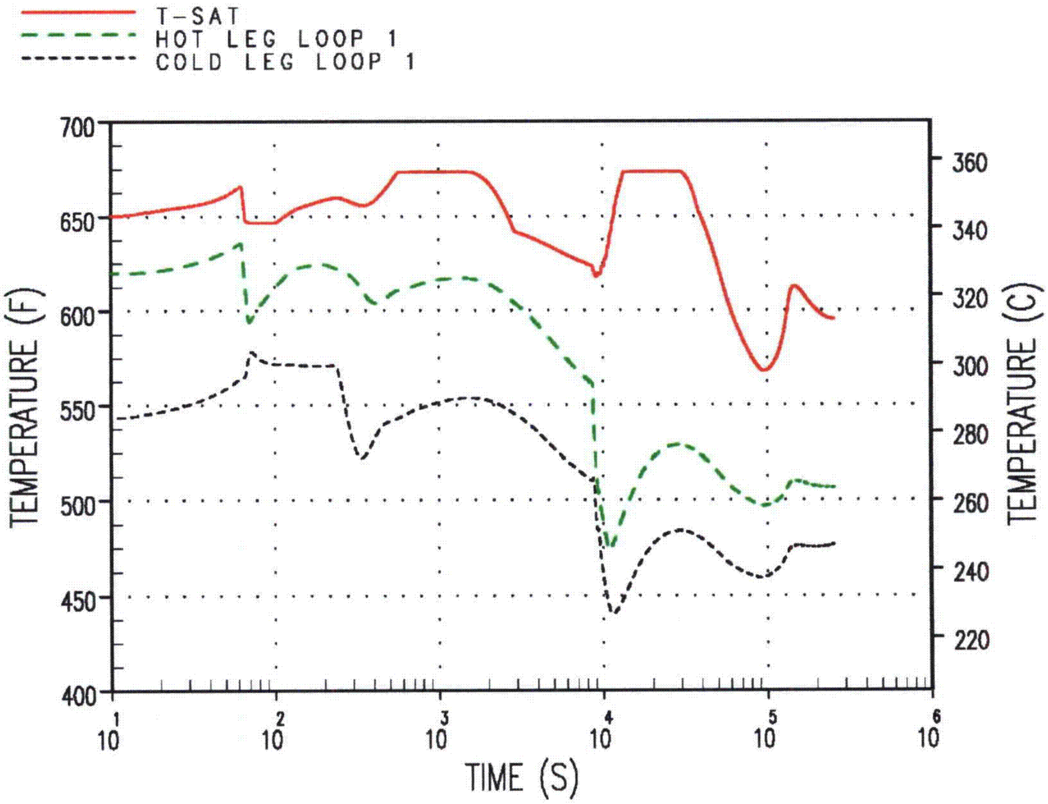

Figure 5: Loop 1 Temperatures (Loop Containing the PRHR-HX)

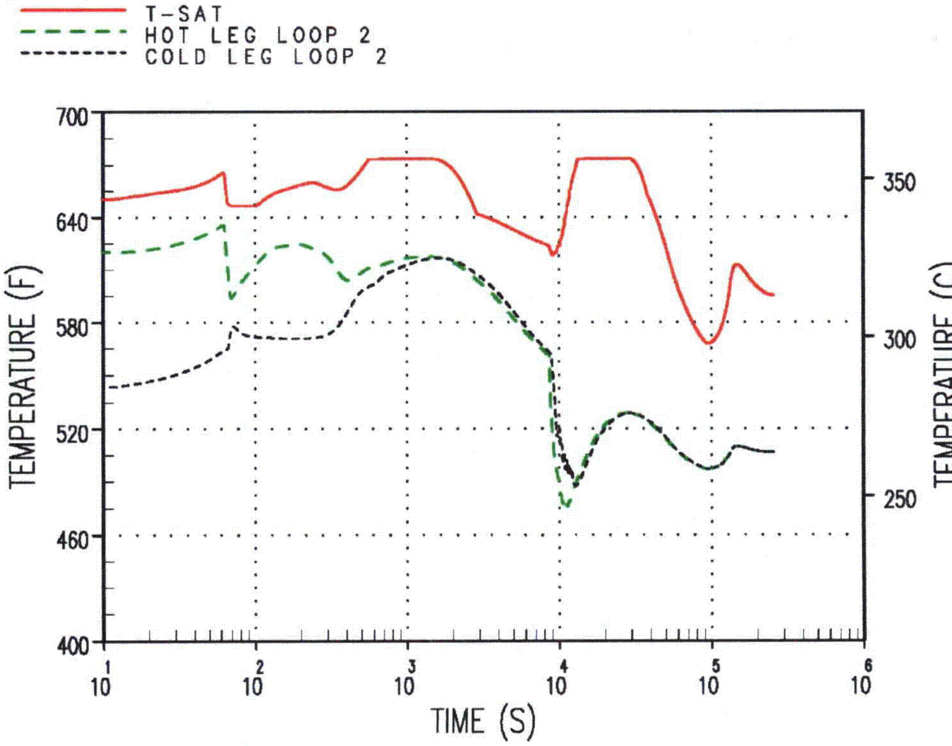

Figure 6: Loop 2 Temperatures (Loop Not Containing the PRHR-HX)

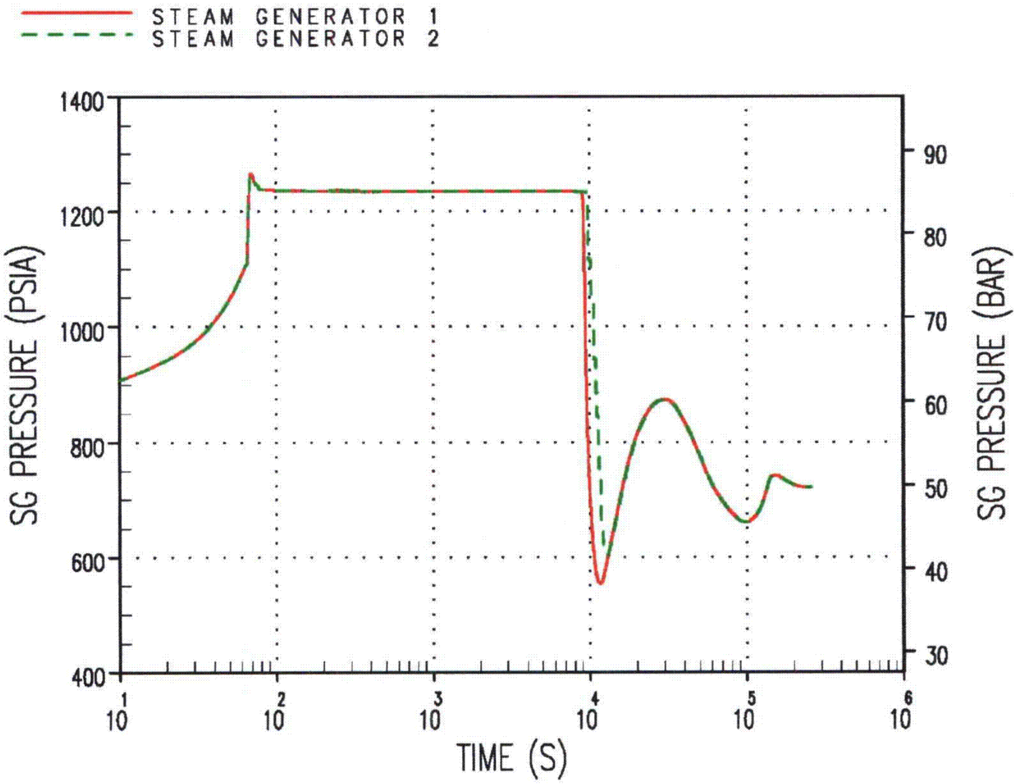

Figure 7: Steam Generator Pressure

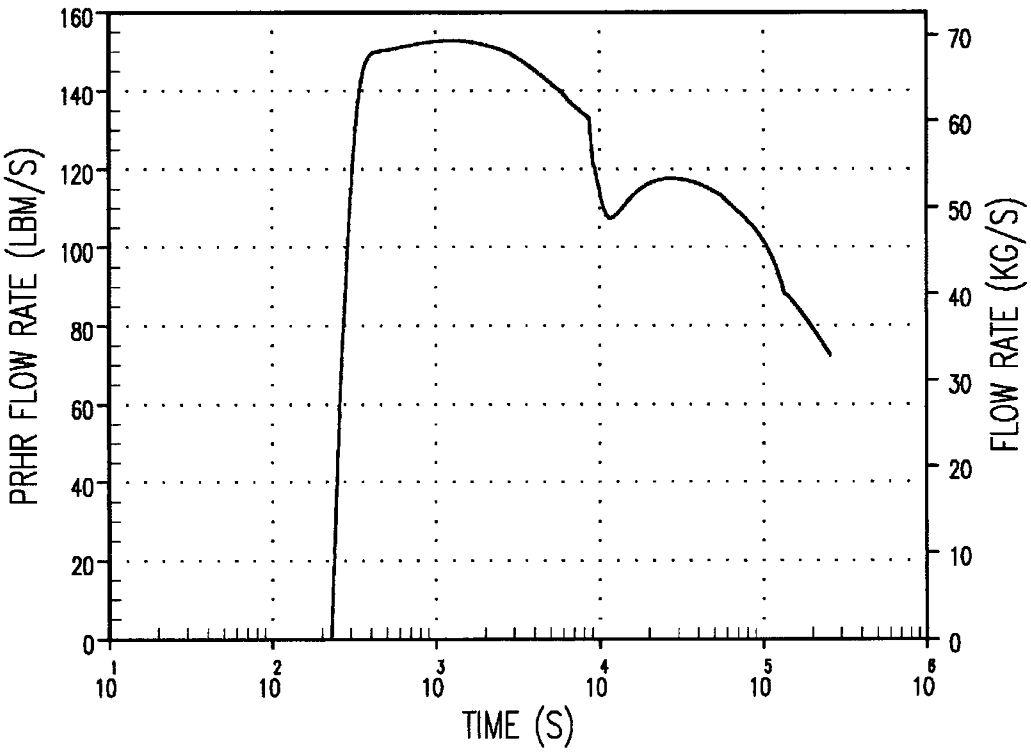

Figure 8: PRHR Flow Rate

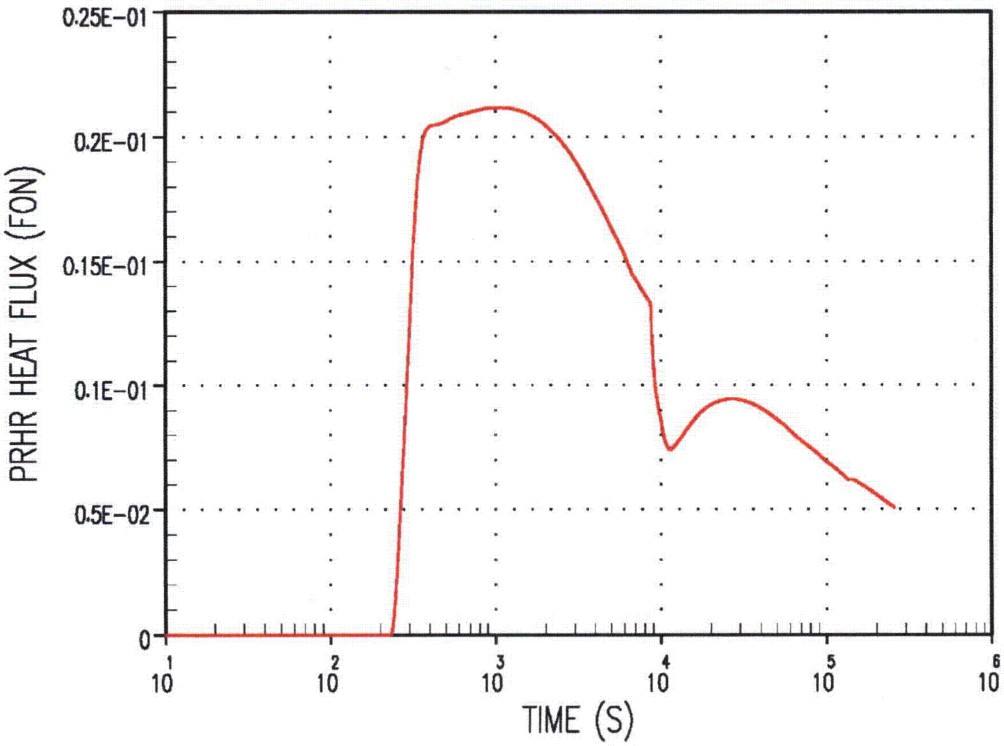

Figure 9: PRHR Heat Flux

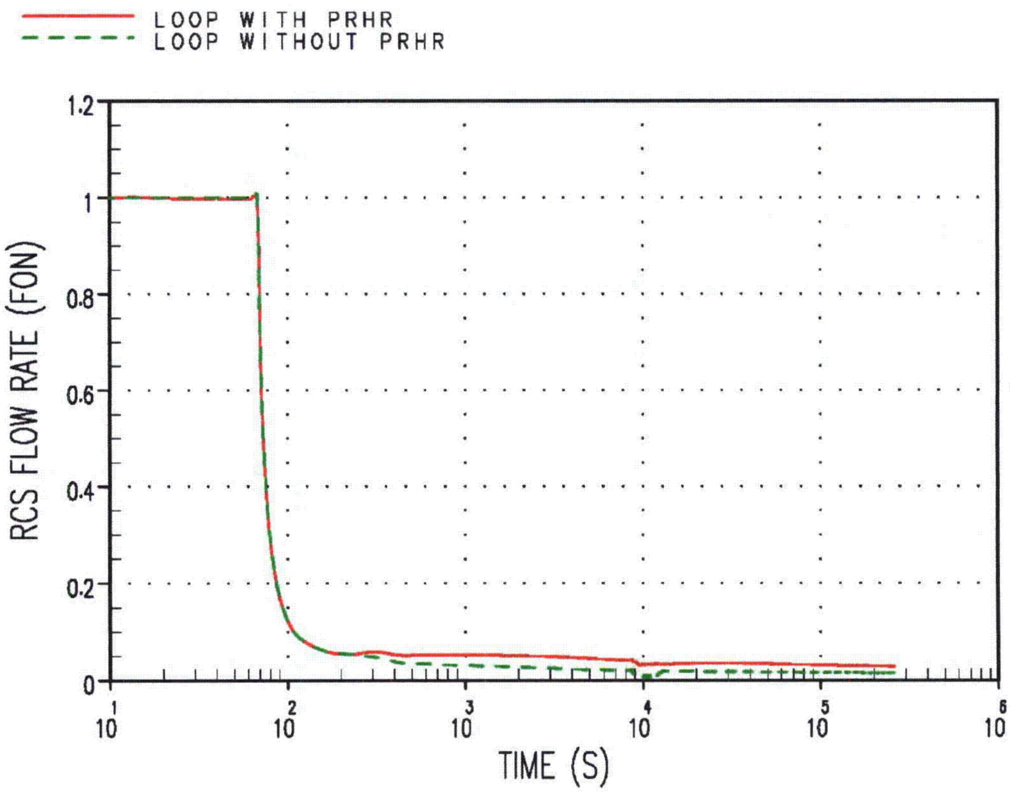

Figure 10: Reactor Coolant Flow Rate

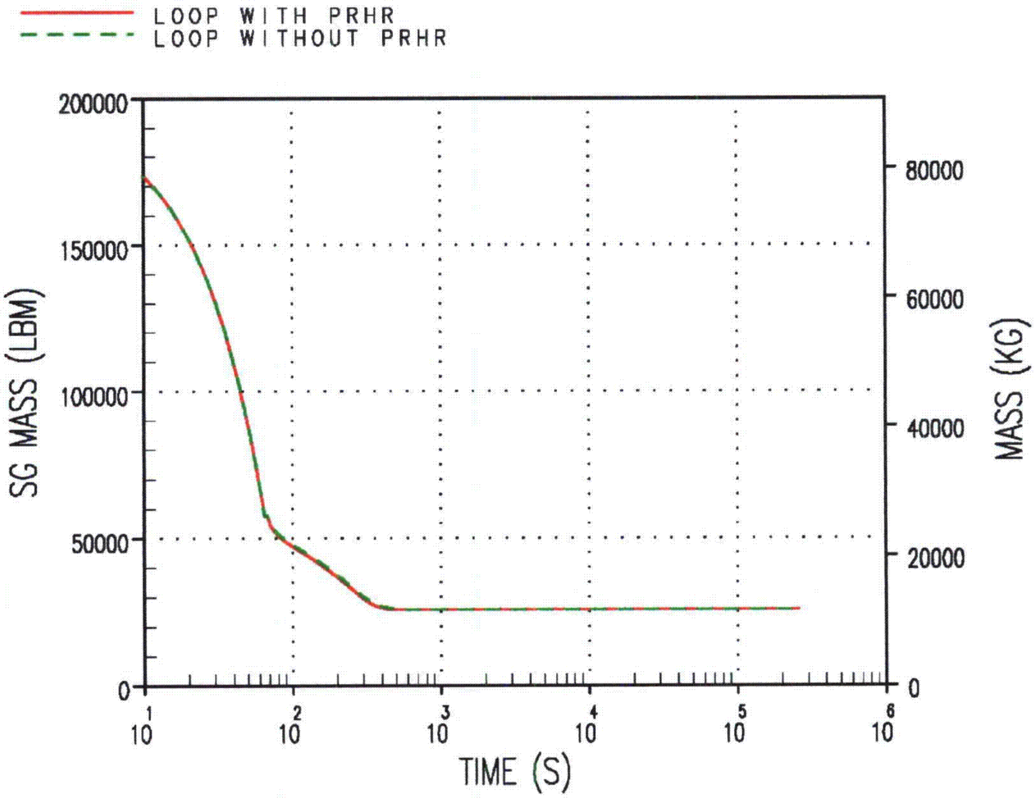

Figure 11: Steam Generator Inventory

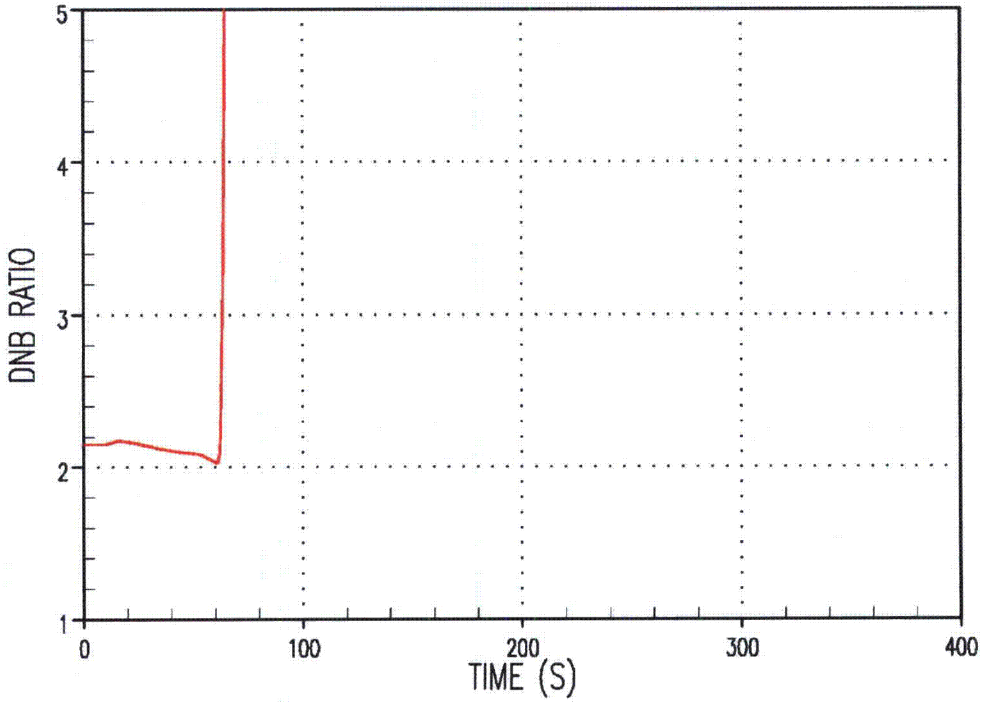

Figure 12: DNB Ratio

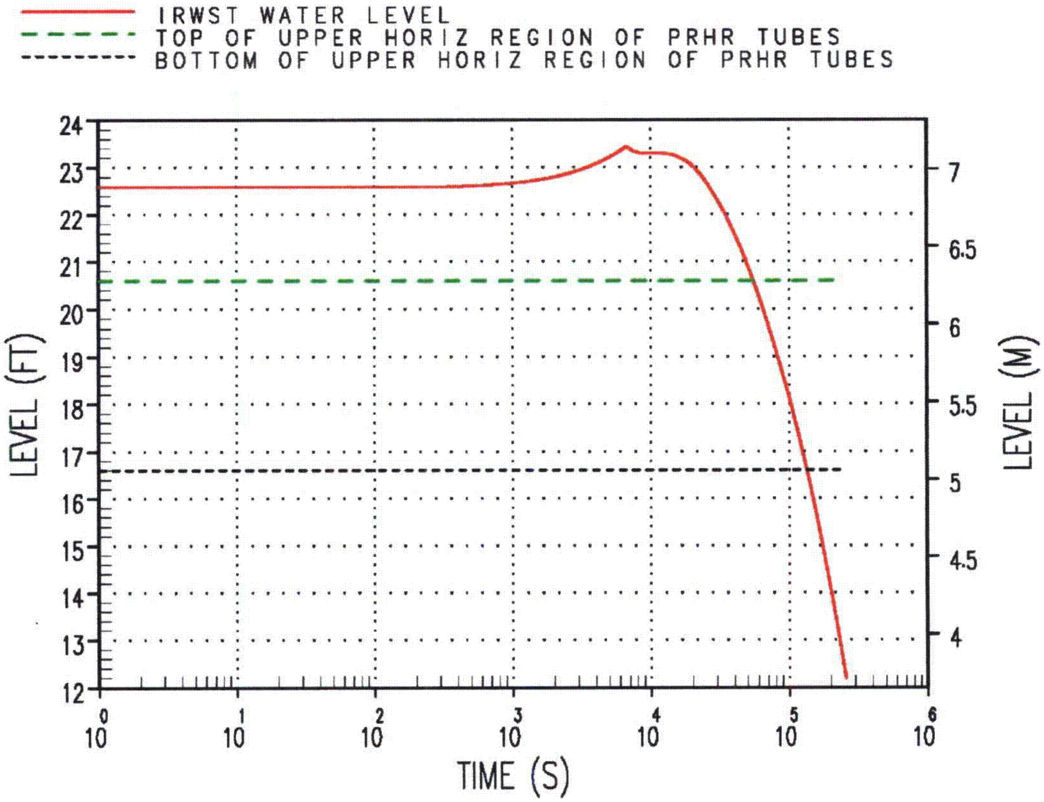

Figure 13: IRWST Water Level*

*Note: the analysis conservatively reduced the initial IRWST water mass [

]a,c

[ ]a,c

Figure 14: Condensate Return (amount returned compared to steamed off)

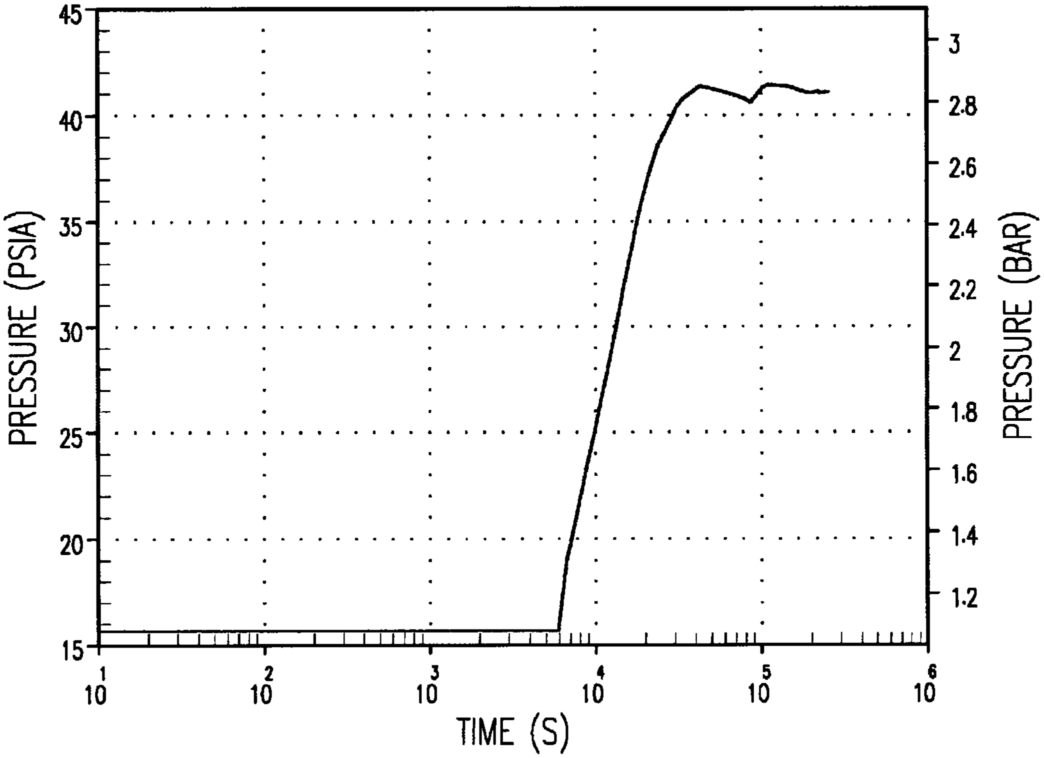

Figure 15: Containment Pressure

Appendix A to 15.02.06-2: Thermal Hydraulic Evaluation of Limiting Event

To confirm the limiting scenario with respect to the passive residual heat removal heat exchanger (PRHR-HX) performance, the events identified for further evaluation were the loss of normal feedwater (LONF) coincident with loss of ac power (LOAC) (loss of ac power to the station auxiliaries), the main steam line break (SLB), and the feedwater line break (FLB) events. These events were further evaluated using the MAAP4.0.7 code [

]a,c

It is noted that MAAP4 is a best estimate code typically used for severe accident analysis. MAAP4 was benchmarked and showed reasonable agreement with the WGOTHIC Containment Analysis condensate return calculation. The integrated core, primary system, secondary system, containment and passive safety system modeling aspect of MAAP4 is useful for drawing insights in the analysis of PXS condensate return as many of the important phenomena are coupled (for example heat transfer from the PRHR and heat losses from the reactor vessel to the containment water). In these cases, the role of MAAP4 is limited to being a screening tool to justify the selection of the limiting scenario for PXS condensate return analysis.

The results of the MAAP4 analyses show that the loss of normal feedwater case with no secondary side break gives the most conservative IRWST water level reduction rate for the PRHR-HX performance. However, the differences between the scenarios are very small and over 72 hours the IRWST water level difference is less than ½ ft (15 cm). The differences between the scenarios occur early in the transients and the behavior over the long term between the cases is practically identical. Key event timing for the cases is presented in Table A.1.

The containment pressure results are presented in Figure A.2. [

]a,c

The cooldown of the RCS during the blowdown of the SGs in the break cases is considerably greater than in the LONF case. Additionally, the CMTs actuate earlier and cool the RCS down earlier in the break cases (Table A.1) than in the LONF case. All cases have reactor trip and actuation of the PRHR-HX within one minute of the event initiation. The PRHR flow is initially forced through the heat exchanger as the RCPs coast down. Then the heat transfer rate drops rapidly as the flow transitions to natural circulation. Because of the energy lost from the primary and secondary systems during the blowdown in the break cases, the initial PRHR heat transfer to the IRWST in the LONF case is considerably higher than in the break cases.

In the secondary side break cases, more of the energy from the primary/secondary system is transferred directly to the passive containment cooling system (PCS) and to the passive heat sinks without heating up or boiling away IRWST water (Figure A.3). [

]a,c In the long term the containment pressure, condensate return, and the PRHR-HX heat removal are approximately the same as seen in Figures A.2, A.5 and A.6, respectively, for all the scenarios. The long term water level in the IRWST is the determining factor in evaluating the limiting scenario (Figure A.7). Because the IRWST water level is slightly lower in the LONF case, the LONF case is considered to be the limiting case for the PRHR-HX performance.

Table A.1: Key Event Timing for Nominal PCS Water Flow Cases

Event	Loss of Feed/SBO	MSLB	FLB	Small FLB
Reactor Trip	41.5 s on Low-SG	1.1 s on RCS ΔT	6.0 s on Hi-Cont P	31.5 s on Hi Cont-P
PRHR Actuated	46.7 s	2.6 s	6.3 s	32.8 s
RCP Trip	42.5 s	2.4 s	6.0 s	31.5 s
CMTs Actuated	7720 s	2.4 s	6.0 s	31.5 s
PCS Actuated	21600 s	2.4 s	6.0 s	31.5 s
IRWST Steaming	15000 s	17200 s	16600 s	17000 s
Top of PRHR-HX Tube Uncovery	32.4 hrs	36.9 hrs	36.0 hrs	36.6 hrs
Peak Cont. Pressure	2.0 bar (29 psia) @ 51500 s	3.4 bar (49 psia) @ 130 s	2.3 bar (33 psia) @ 180 s	2.0 bar (29 psia) @ 50400 s

a,c

Figure A.1

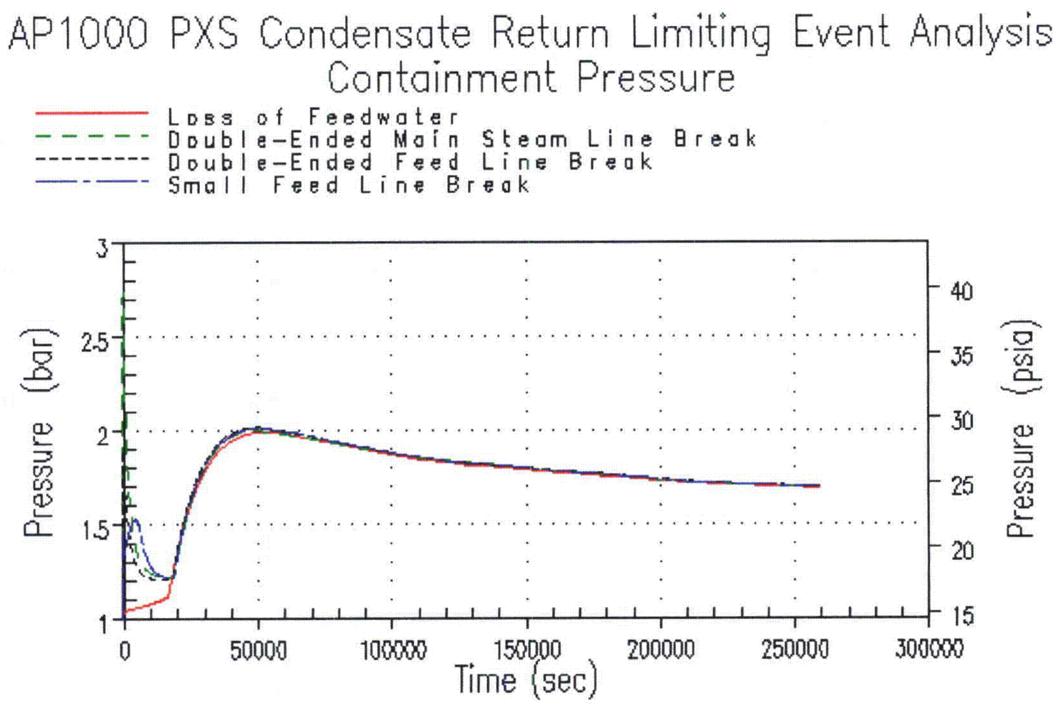

Figure A.2

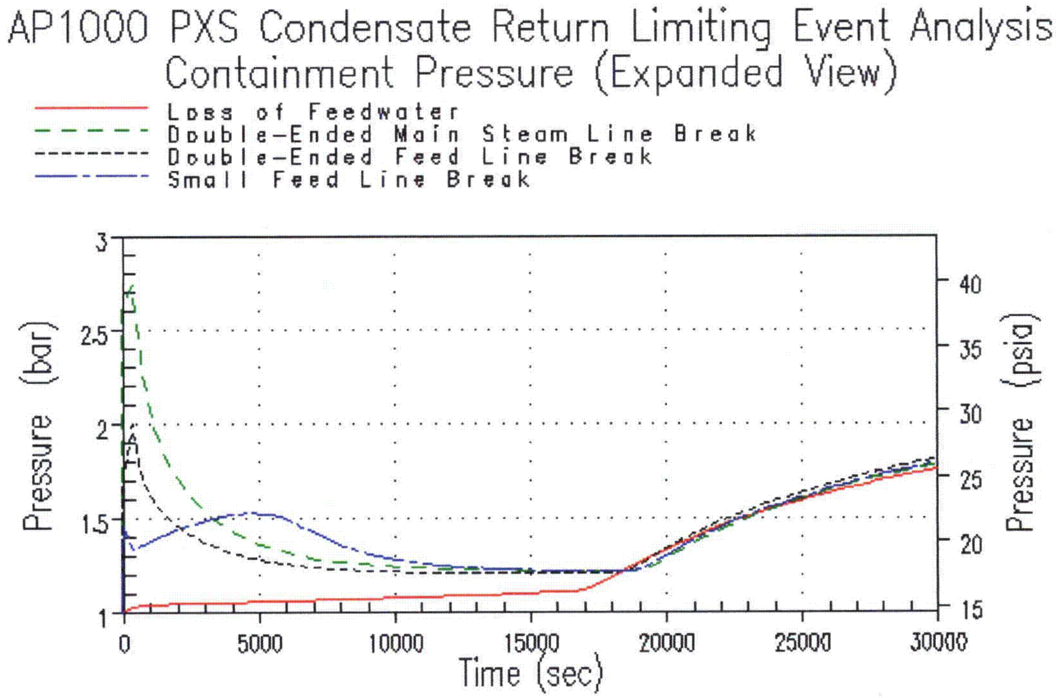

Figure A.2a

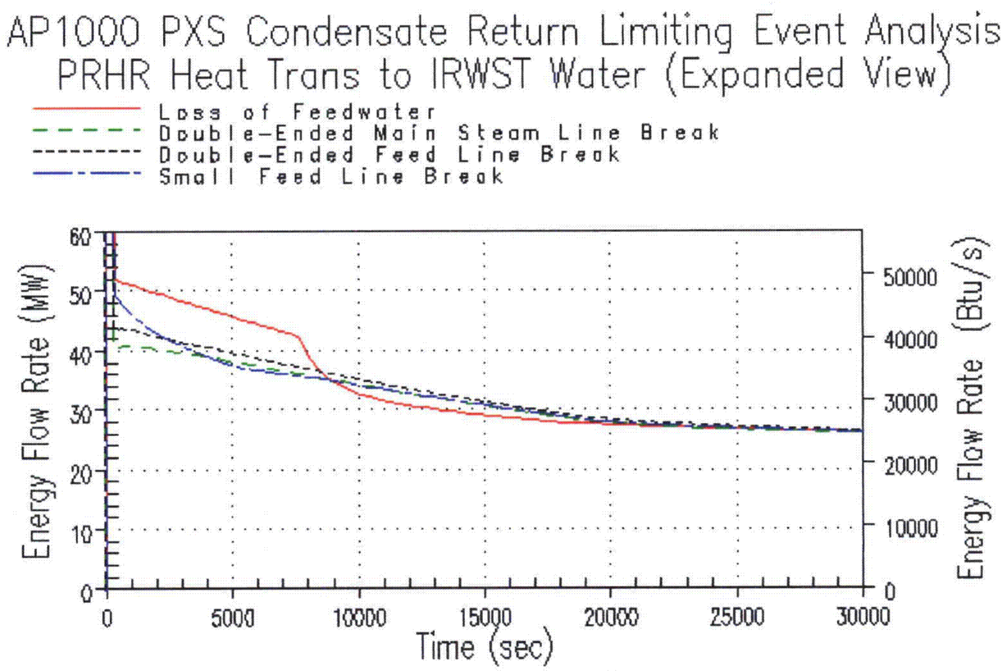

Figure A.3

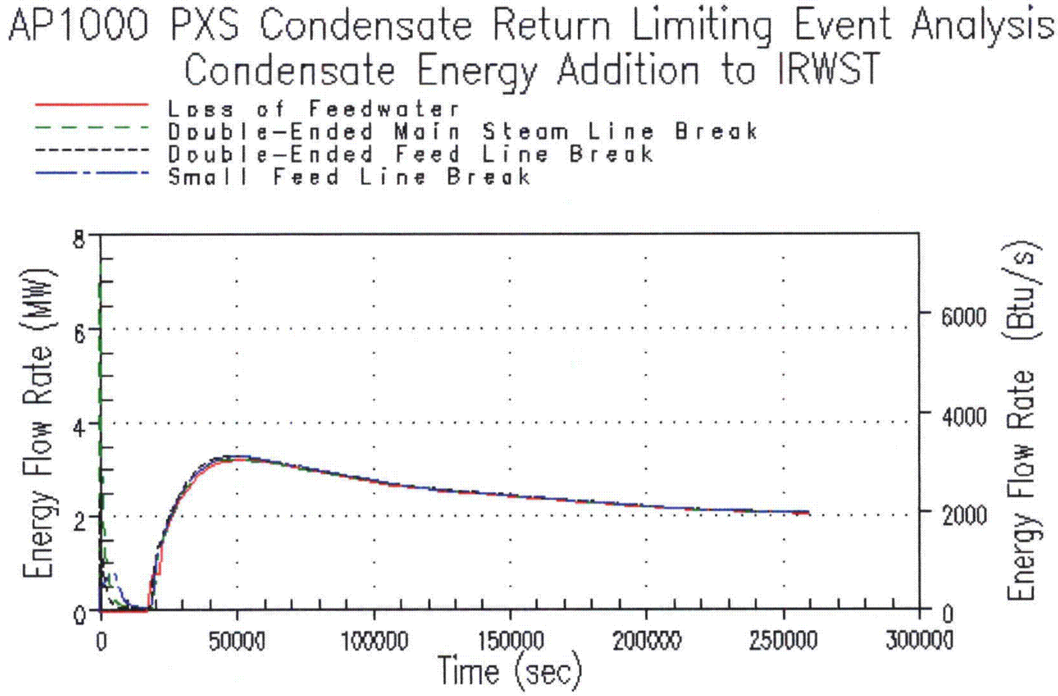

Figure A.4

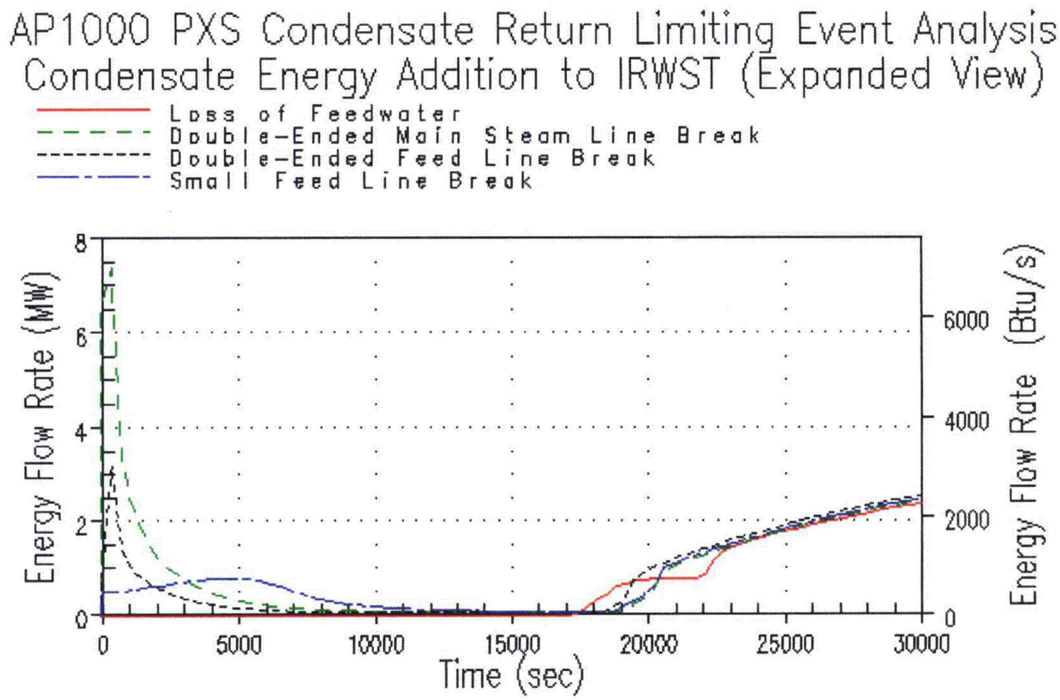

Figure A.4a

a,c

Figure A.5

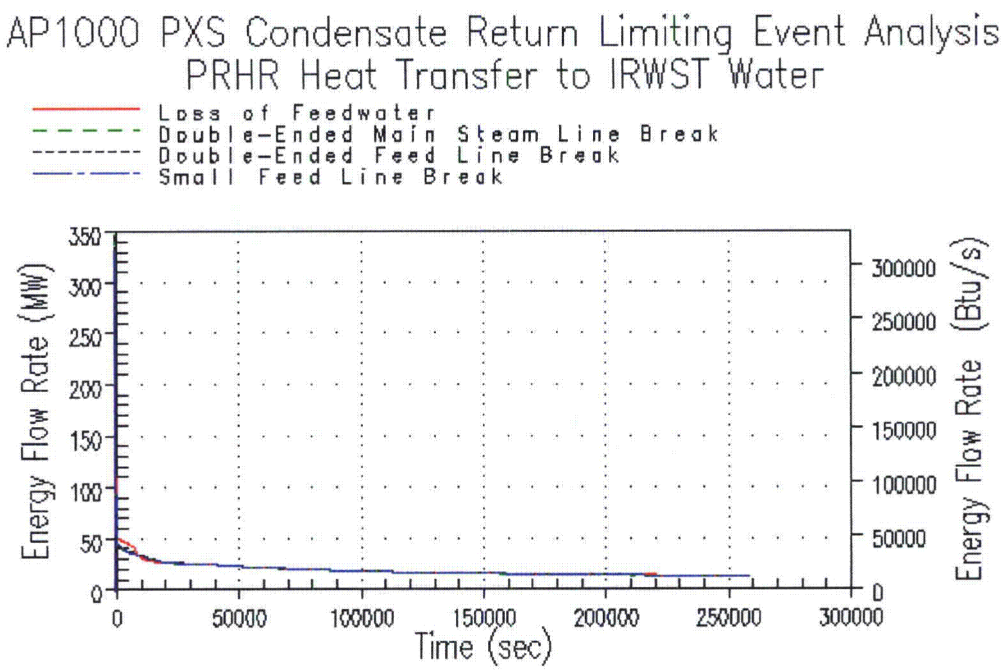

Figure A.6

Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return

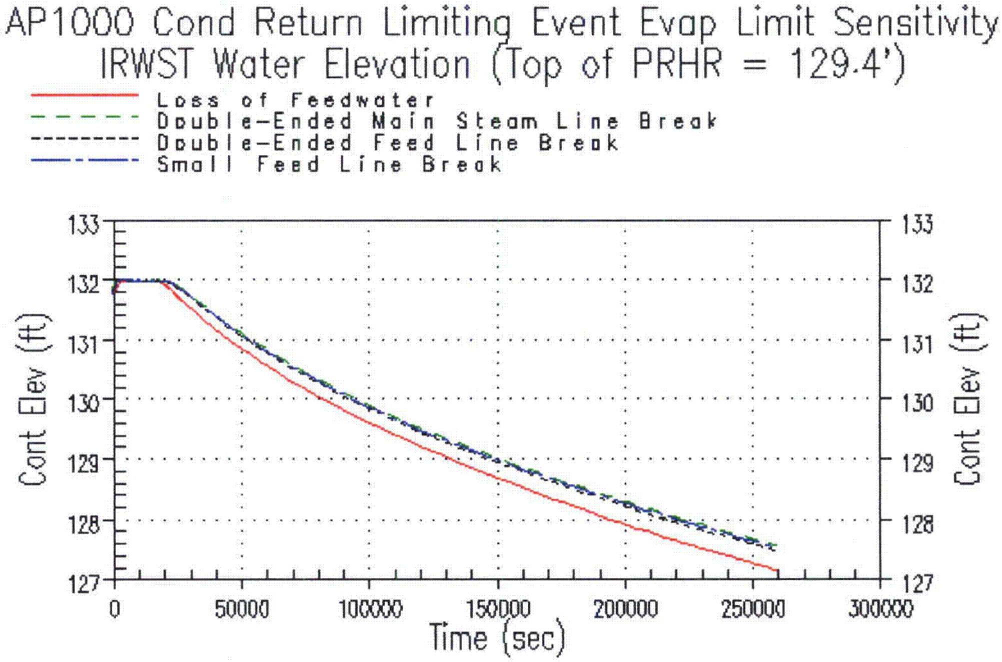

Figure A.7

Enclosure 2
Westinghouse Application Letter CAW-14-3961
and Affidavit
(7 pages including cover page)

Westinghouse

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Project letter: APC_APG_000144

Our ref: CAW-14-3961

June 19, 2014

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: **Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return** (Proprietary) and (Non-Proprietary)

The proprietary information for which withholding is being requested in the above-referenced letter is further identified in the affidavit signed by Westinghouse Electric Company LLC. The affidavit accompanying this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and address with specificity the considerations listed in paragraph (b) (4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by APOG.

Correspondence with respect to the proprietary aspects of this application for withholding or the accompanying affidavit should reference CAW-14-3961 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

Hank A. Sepp
Director, MCRE, MCRE-Engineering Services

CAW-14-3961
June 19, 2014

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared **Hank A. Sepp**, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Hank A. Sepp
Director, MCRE, MCRE-Engineering Services

Sworn to and subscribed
before me this 9th day
of June 2014.

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Linda J. Bugle, Notary Public
City of Pittsburgh, Allegheny County
My Commission Expires June 18, 2017
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

Linda J. Bugle
Notary Public

(1) I am Director, MCRE, MCRE-Engineering Services, Westinghouse Electric Company, LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390; it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld from within **Responses to NRC RAIs 06.03-5 and 15.02.06-2 on Condensate Return**, and may be used only for that purpose.

The information requested to be withheld reveals details of the AP1000 design; sequence and method of construction; and timing and content of inspection and testing. This information was developed and continues to be developed by Westinghouse. The information is part of that which enables Westinghouse to manufacture and deliver products to utilities based on proprietary designs.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar commercial power reactors without commensurate expenses.

The information requested to be withheld is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Enclosure 3
Proprietary Information Notice and Copyright Notice
(2 pages including cover page)

June 19, 2014

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Duke Energy

Enclosure 4

Levy Nuclear Plant Units 1 and 2

AP1000 DCD Tier 2 Licensing Basis Document - Proposed Changes (14 pages including cover page)

AP1000 DCD Tier 2 Licensing Basis – Proposed Changes:

5.4.14.1 Design Bases

The passive residual heat removal heat exchanger automatically actuates to remove core decay heat for an extended period of time as discussed in Section 6.3, assuming the condensate from steam generated in the in-containment refueling water storage tank (IRWST) is returned to the tank. The passive residual heat removal heat exchanger is designed to withstand the design environment of 2500 psia and 650°F.

The passive residual heat removal heat exchanger and the in-containment refueling water storage tank are designed to delay significant steam release to the containment for at least one hour. The passive residual heat removal heat exchanger will remove sufficient decay heat from the reactor coolant system to satisfy the applicable post-accident safety evaluation criteria detailed in Chapter 15. In addition, the passive residual heat removal heat exchanger will cool the reactor coolant system, with reactor coolant pumps operating or in the natural circulation mode, so that the reactor coolant system pressure can be lowered to reduce stress levels in the system if required. See Section 6.3 for a discussion of the capability of the passive core cooling system.

The passive residual heat removal heat exchanger is designed and fabricated according to the ASME Code, Section III, as a Class 1 component. Those portions of the passive residual heat exchanger that support the primary-side pressure boundary and falls under the jurisdiction of ASME Code, Section III, Subsection NF are AP1000 equipment Class A (ANS Safety Class 1, Quality Group A). Stresses for ASME Code, Section III equipment and supports are maintained within the limits of Section III of the Code. Section 5.2 provides ASME Code, Section III and material requirements. Subsection 5.2.4 discusses inservice inspection.

Materials of construction are specified to minimize corrosion/erosion and to provide compatibility with the operating environment, including the expected radiation level. Subsection 5.2.3 discusses the welding, cutting, heat treating and other processes used to minimize sensitization of stainless steel.

Chapter 6 Table of Contents (actual page number changes are not shown)

6.3.1.1.1 Emergency Core Decay Heat Removal

For postulated non-LOCA events, where a loss of capability to remove core decay heat via the steam generators occurs, the passive core cooling system is designed to perform the following functions:

- The passive residual heat removal heat exchanger automatically actuates to provide reactor coolant system cooling.
- The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate collection features and the passive containment cooling system, are designed to remove decay heat following a design basis event. Automatic depressurization actuation is not expected; but may occur depending on the amount of reactor coolant system leakage and when normal systems are recovered (refer to subsection 6.3.1.1.4).
- The passive residual heat removal heat exchanger is designed to maintain acceptable reactor coolant system conditions for at least 72 hours following a non-LOCA event. The applicable post-accident safety evaluation criteria are discussed in Chapter 15. Operator action may be taken in accordance with emergency procedures to de-energize the loads on the Class 1E batteries to avoid unnecessary automatic actuation of the automatic depressurization system. Specific safe shutdown criteria are described in subsection 6.3.1.1.4.
- The passive residual heat removal heat exchanger is capable of performing its post-accident safety functions, assuming the steam generated in the in-containment refueling water storage tank is condensed on the containment vessel and returned by gravity via the in-containment refueling water storage tank condensate return gutter and downspouts.
- During a steam generator tube rupture event, the passive residual heat removal heat exchanger removes core decay heat and reduces reactor coolant system temperature and pressure, equalizing with steam generator pressure and terminating break flow, without overfilling the steam generator.

6.3.1.1.4 Safe Shutdown

The functional requirements for the passive core cooling system specify that the plant be brought to a stable condition using the passive residual heat removal heat exchanger for events not involving a loss of coolant. As stated in subsection 6.3.1.1.1, the passive residual heat removal heat exchanger in conjunction with the passive containment cooling system provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. Additionally, the passive core cooling system, in conjunction with the passive containment cooling system, has diverse capability to establish long-term safe shutdown conditions in the reactor coolant system, eventually cooling the reactor coolant system to about 420°F in 36 hours, with or without availability of the reactor coolant pumps.

The core makeup tanks automatically provide injection to the reactor coolant system after they are actuated on low reactor coolant temperature or low pressurizer pressure or level. The passive core cooling system can maintain stable plant conditions for a long time in this mode of operation, depending on the reactor coolant leakage and the availability of ac power sources. For example, with a technical specification leak rate of 10 gpm, stable plant conditions can be maintained for at least 10 hours. With a smaller leak a longer time is available.

In most sequences the operators would return the plant to normal system operations and terminate passive system operation in accordance with the plant emergency operating procedures. In scenarios when ac power sources are unavailable for approximately 22 hours, the automatic depressurization system will automatically actuate. However, after initial plant cooldown following a non-LOCA event, operators will assess plant conditions and have the option to perform recovery actions to further cool and depressurize the reactor coolant system in a closed-loop mode of operation, i.e., without actuation of the automatic depressurization system. After verifying the reactor coolant system is in an acceptable, stable condition such that automatic depressurization is not needed, the operators may take action to extend passive residual heat removal heat exchanger operation by de-energizing the loads on the Class 1E dc batteries powering the protection and monitoring system actuation cabinets. After operators have taken action to extend its operation, the passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, has the capability to establish and maintain long-term safe shutdown conditions. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time.

For loss of coolant accidents, when the core makeup tank level reaches the automatic depressurization system actuation setpoint and other postulated events where ac power sources are lost but passive residual heat removal heat exchanger operation is not extended or is exhausted, the automatic depressurization system will be initiated. This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once the reactor coolant system is nearly depressurized. For these conditions, the reactor coolant system depressurizes to saturated conditions at about 250°F within 24 hours. The passive core cooling system can maintain this safe shutdown condition indefinitely for the plant.

The basis used to define the passive core cooling system functional requirements is derived from Section 7.4 of the Standard Review Plan. The functional requirements are met over the range of anticipated events and single failure assumptions. The primary function of the passive core cooling system during a safe shutdown using only safety-related equipment is to provide a means for boration, injection, and core cooling. Details of the safe shutdown design bases are presented in subsection 5.4.7 and Section 7.4. The performance of the passive residual heat removal heat exchanger to bring the plant to 420°F in 36 hours is summarized in subsection 19E.4.10.2.

6.3.1.2 Nonsafety Design Basis

6.3.1.2.1 Post-Accident Core Decay Heat Removal

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features and the passive containment cooling system, has the capability to maintain the reactor coolant system in the specified, long-term safe shutdown condition for 14 days in a closed-loop mode of operation. The automatic depressurization system can be manually actuated by the operators at any time during extended passive residual heat removal heat exchanger operation to initiate open-loop cooling. The operator actions necessary to achieve safe shutdown using the passive residual heat removal heat exchanger in a closed-loop mode of operation involve preventing unnecessary actuation of the automatic depressurization system as detailed in subsection 7.4.1.1.

6.3.1.3 Power Generation Design Basis

6.3.2.1.1 Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions

For events not involving a loss of coolant, the emergency core decay heat removal is provided by the passive core cooling system via the passive residual heat removal heat exchanger. The heat exchanger consists of a bank of C-tubes, connected to a tubesheet and channel head arrangement at the top (inlet) and bottom (outlet). The passive residual heat removal heat exchanger connects to the reactor coolant system through an inlet line from one reactor coolant system hot leg (through a tee from one of the fourth stage automatic depressurization lines) and an outlet line to the associated steam generator cold leg plenum (reactor coolant pump suction).

The inlet line is normally open and connects to the upper passive residual heat removal heat exchanger channel head. The inlet line is connected to the top of the hot leg and is routed continuously upward to the high point near the heat exchanger inlet. The normal water temperature in the inlet line will be hotter than the discharge line.

The outlet line contains normally closed air-operated valves that open on loss of air pressure or on control signal actuation. The alignment of the passive residual heat removal heat exchanger (with a normally open inlet motor-operated valve and normally closed outlet air-operated valves) maintains the heat exchanger full of reactor coolant at reactor coolant system pressure. The water temperature in the heat exchanger is about the same as the water in the in-containment refueling water storage tank, so that a thermal driving head is established and maintained during plant operation.

The heat exchanger is elevated above the reactor coolant system loops to induce natural circulation flow through the heat exchanger when the reactor coolant pumps are not available. The passive residual heat removal heat exchanger piping arrangement also allows actuation of the heat exchanger with reactor coolant pumps operating. When the reactor coolant pumps are operating, they provide forced flow in the same direction as natural circulation flow through the heat exchanger. If the pumps are operating and subsequently trip, then natural circulation continues to provide the driving head for heat exchanger flow.

The heat exchanger is located in the in-containment refueling water storage tank, which provides the heat sink for the heat exchanger.

Although gas accumulation is not expected, there is a vertical pipe stub on the top of the inlet piping high point that serves as a gas collection chamber. Level detectors indicate when gases have collected in this area. There are provisions to allow the operators to open manual valves to locally vent these gases to the in-containment refueling water storage tank.

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate return features and the passive containment cooling system, can provide core cooling for at least 72 hours. After the in-containment refueling water storage tank water reaches its saturation temperature (in several hours), the process of steaming to the containment initiates. Containment pressure will increase as steam is released from the in-containment refueling water storage tank. As the containment temperature increases, condensation begins to form on the subcooled metal and concrete surfaces inside containment. Condensation on these heat sink surfaces transfers energy to the bulk metal and concrete until they come into equilibrium with the containment atmosphere. Condensation that is not returned to the in-containment refueling water storage tank drains to the containment sump.

Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. Most of the condensate formed on the containment vessel wall is collected in a safety-related gutter arrangement. A gutter is located near the operating deck elevation, and a downspout piping system is connected at the polar crane girder and internal stiffener, to collect steam condensate inside the containment during passive containment cooling system operation and return it to the in-containment refueling water storage tank. The gutter and downspouts normally drain to the containment sump, but when the passive residual heat removal heat exchanger actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the in-containment refueling water storage tank. Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for an extended period of time.

The passive residual heat removal heat exchanger is used to maintain an acceptable, stable reactor coolant system condition. It transfers decay heat and sensible heat from the reactor coolant system to the in-containment refueling water storage tank, the containment atmosphere, the containment vessel, and finally to the ultimate heat sink–the atmosphere outside of containment. This occurs after in-containment refueling water storage tank saturation is reached and steaming to containment initiates.

The duration the passive residual heat removal heat exchanger can continue to remove decay heat is affected by the efficiency of the return of condensate to the in-containment refueling water storage tank. The in-containment refueling water storage tank water level is affected by the amount of steam that leaves the tank and does not return. Offsite or onsite ac power sources are typically recovered within a day, which would allow the operators to place active, defense-in-depth systems into service and to terminate passive system operation. If ac power is not recovered within this time frame, closed-loop cooling using the passive residual heat removal heat exchanger can be extended as described in subsection 7.4.1.1 to maintain a safe, stable condition after a design basis event.

6.3.2.8 Manual Actions

The passive core cooling system is automatically actuated for those events as presented in subsection 6.3.3. Following actuation, the passive core cooling system continues to operate in the injection mode until the transition to recirculation initiates automatically following containment floodup.

Although the passive core cooling system operates automatically, operator actions would be beneficial, in some cases, in reducing the consequences of an event. For example, in a steam generator tube rupture with no operator action, the protection and safety monitoring system automatically terminates the leak, prevents steam generator overfill, and limits the offsite doses. However, the operator can initiate actions, similar to those taken in current plants, to identify and isolate the faulted steam generator, cool down and depressurize the reactor coolant system to terminate the break flow to the steam generator, and stabilize plant conditions.

The operator can take action to avoid actuation of the automatic depressurization system when it is not needed. For non-LOCA events during which ac power has been lost for more than 22 hours, the protection and safety monitoring system will automatically open the automatic depressurization system valves to begin a controlled depressurization of the reactor coolant system and, eventually, containment floodup and recirculation prior to depletion of the actuation batteries. However, the operators can take action to block actuation of the automatic depressurization system should actuation be deemed unnecessary based on reactor coolant system conditions. This action allows closed loop passive residual heat removal heat exchanger operation to continue as long as acceptable reactor coolant system conditions are maintained.

Section 7.4 describes the anticipated operator actions to block unnecessary automatic depressurization system actuation. Section 7.5 describes the post-accident monitoring instrumentation available to the operator in the main control room following an event.

The following, highlighted text will be added to subsection 6.3.3, "Performance Evaluation."

B. Decrease in heat removal by the secondary system

1. Loss of Main Feedwater Flow

2. Feedwater system piping failure

...

For non-LOCA events, the passive residual heat removal heat exchanger is actuated so that it can remove core decay heat. The passive residual heat removal heat exchanger can operate for at least 72 hours after initiation of a design basis event to satisfy Condition I, II, III, and IV safety evaluation criteria described in the relevant safety analyses. Subsection 6.3.3.2.1 provides an evaluation of the duration of the passive residual heat removal heat exchanger operation using the LOFTRAN code described in subsection 15.0.11.2. In this evaluation it is assumed that the operators power down the protection and monitoring actuation cabinets in the 22 hour time frame prior to the automatic timer actuating ADS.

In addition to mitigating the initiating events, the passive residual heat removal heat exchanger is capable of cooling the reactor coolant system to the specified safe shutdown condition as described in subsection 19E.4.10.2. A non-bounding, conservative estimation of the plant response during operator-initiated, extended operation of the passive residual heat removal heat exchanger is demonstrated in the shutdown temperature evaluation of subsection 19E.4.10.2. The closed-loop cooling mode allows the reactor coolant system pressure to decrease and reduces the stress in the reactor coolant system and connecting pipe to low levels. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.

...

As the core makeup tanks drain down, the automatic depressurization system valves are sequentially actuated. The depressurization sequence establishes reactor coolant pressure conditions that allow injection from the accumulators, and then from the in-containment refueling water storage tank and the containment recirculation path. Therefore, an injection source is continually available. If onsite or offsite ac power has not been restored after 72 hours, the post-72 hour support actions described in subsection 1.9.5.4 maintain this mode of core cooling and provide adequate decay heat removal for an unlimited time.

The transient analyses summarized in Chapter 15 are extended long enough to demonstrate the applicable safety evaluation criteria are met. It is expected that normal systems would be available such that operators could terminate the passive safety systems and proceed with an orderly shutdown. However, as discussed in subsection 6.3.1.1.4, the passive systems are capable of bring the plant to a safe shutdown condition and maintaining that condition.

6.3.3.2.1 Loss of Main Feedwater

The most severe core conditions resulting from a loss of main feedwater system flow are associated with a loss of flow at full power. The heat-up transient effects of loss of flow at reduced power levels are bounded by the loss of flow at full power. Subsection 15.2.7 provides a description of this event, including criteria and analytical results.

For this event, the passive residual heat removal heat exchanger is actuated. If the core makeup tanks are not initially actuated, they actuate later when passive residual heat exchanger cooling sufficiently reduces pressurizer level. The passive residual heat removal heat exchanger serves to remove core decay heat and the core makeup tanks inject a borated water solution directly into the reactor vessel downcomer annulus. Since the reactor coolant pumps are tripped on actuation of the core makeup tanks, the passive residual heat removal heat exchanger operates under natural circulation conditions. The core makeup tanks operate via water recirculation, without draining, to maintain reactor coolant system inventory. Therefore, the automatic depressurization system is not actuated on the lowering of the core makeup tank level. Since the event is characterized by a heat-up transient, the injection of negative reactivity is not required and is not taken credit for in the analysis to control core reactivity.

The reactor coolant system does not depressurize to permit the accumulators to deliver makeup water to the reactor coolant system. Subsequent to stabilizing plant conditions and satisfying passive core cooling system termination criteria, the operator terminates passive core cooling system operation and initiates a normal plant shutdown.

6.3.3.2.1.1 Loss of AC Power to the Plant Auxiliaries

The most severe conditions resulting from a loss of ac power to the plant auxiliaries are associated with loss of offsite power with a loss of main feedwater system flow at full power. A loss of main feedwater with a loss of ac power lasting longer than a few hours presents the highest demand on passive residual heat removal heat exchanger operation. Subsection 15.2.6 provides a description of this short-term event, including criteria and analytical results.

During most events, the passive systems would be terminated in hours. However, if normal systems are not recovered as expected, the passive residual heat removal heat exchanger removes core decay heat and maintains acceptable reactor coolant system conditions for at least 72 hours. For a non-loss of coolant accident event lasting as long as 24 hours, the automatic depressurization system will actuate if operators do not act to avoid actuation when it is not needed. For this long-term transient, it is assumed operators extend passive residual heat removal heat exchanger operation as described in subsection 7.4.1.1, such that the automatic depressurization system does not actuate.

The loss of main feedwater with loss of ac power event is analyzed for a 72 hour period, assuming operators extend closed-loop cooling beyond the time the automatic depressurization system would be actuated by the protection and safety monitoring system. This event mirrors the loss of ac power to the plant auxiliaries event described in subsection 15.2.6, but the loss of ac power extends to 72 hours. In this event, operation of the passive residual heat removal heat exchanger continues for 72 hours and maintains acceptable reactor coolant system conditions such that the applicable Condition II safety evaluation criteria are met.

Reactor coolant system leakage could limit closed-loop capacity. A reactor coolant system leak could produce conditions that would preclude the operators from de-energizing the loads on the Class 1E batteries, or could require the operators to re-energize the buses powered by the Class 1E batteries before 72 hours so that the automatic depressurization system valves could be actuated. When an ac power source is restored and passive core cooling system termination criteria are satisfied, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

6.3.3.4.1 Loss of Startup Feedwater During Hot Standby, Cooldowns, and Heat-ups

...

The in-containment refueling water storage tank provides the heat sink for the passive residual heat removal heat exchanger. Initially, the heat addition increases the water temperature. Within one to two hours, the water reaches saturation temperature and begins to boil. The steam generated in the in-containment refueling water storage tank discharges to containment. Because the containment integrity is maintained during cooldown Modes 3 and 4, the passive containment cooling system provides the safety-related ultimate heat sink. Therefore, most of the steam generated in the in-containment refueling water storage tank is condensed on the inside of the containment vessel and drains back into the in-containment refueling water storage tank via the condensate return gutter arrangement. This allows it to function as a heat sink.

7.4.1.1 Safe Shutdown Using Safety-Related Systems

The engineered safety system actuation signal generated on low pressurizer pressure also actuates containment isolation. This prevents loss of water inventory from containment and permits extended operation of the passive residual heat removal heat exchanger and the in-containment refueling water storage tank.

...

A gutter located at the operating deck elevation collects condensate from the inside of the containment shell. Valves located in drain lines from the gutter to the containment waste sump close on a passive residual heat removal heat exchanger actuation signal. This action diverts the condensate to the in-containment refueling water storage tank. The system provides core decay heat removal in this configuration without an increase in the containment water level.

Once the reactor coolant system and the safety systems are in this configuration, the plant is in a stable shutdown condition. The reactor coolant system temperatures and pressures continue to slowly decrease. The passive residual heat removal heat exchanger has the capacity to maintain a safe, stable reactor coolant system condition during a design basis event for at least 72 hours in a closed-loop mode of operation. A non-bounding, conservative analysis of extended operation in this mode shows the passive residual heat removal heat exchanger cools the reactor coolant system to 420°F in 36 hours.

Operation in this configuration may be limited in time duration by reactor coolant system leakage. The core makeup tanks can only supply a limited amount of makeup in the event there is reactor coolant system leakage. Eventually the volume of the water in the core makeup tanks will decrease to the first stage automatic depressurization setpoint. The time to reach this setpoint depends upon the reactor coolant system leak rate and the reactor coolant cooldown.

The Class 1E dc batteries that power the automatic depressurization system valves provide power for at least 24 hours. There is a timer that measures the time that ac power sources are unavailable. This timer provides for automatic actuation of the automatic depressurization system before the Class 1E dc batteries are discharged. The emergency response guidelines direct the operator to assess the need for automatic depressurization before the timer completes its count (approximately 22 hours). The operator assessment includes consideration for a visible refueling water storage tank level, full core makeup tanks, a high and stable pressurizer level, and decreasing or stable reactor coolant system temperature. If automatic depressurization is not needed, the operator is directed to de-energize all loads on the Class 1E dc batteries. This action preserves the capability for the operator to initiate automatic depressurization at a later time based on assessment of the same parameters.

The following change would be made on Sheet 11 of Table 9.5.1-1, "AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1":

73. Fire damage should be limited so that systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station can be repaired within 72 hours.	C.5.b (1)	AC	Safe shutdown following a fire is defined for the AP1000 plant as the ability to achieve and maintain the reactor coolant system (RCS) temperature below 215.6°C (420°F) without uncontrolled venting of the primary coolant from the RCS. This is a departure from the criteria applied to the evolutionary plant designs, and the existing plants where safe shutdown for fires applies to both hot and cold shutdown capability. With expected RCS leakage, the AP1000 plant can maintain safe shutdown conditions for at least 14 days. Therefore, repairs to systems necessary to reach cold shutdown need not be completed within 72 hours.

15.0.13 Operator Actions

For events where the PRHR heat exchanger is actuated, the plant automatically cools down to a safe, stable condition. Where a stabilized condition is reached automatically following a reactor trip, it is expected that the operator may, following event recognition, take manual control and proceed with orderly shutdown of the reactor in accordance with the normal, abnormal, or emergency operating procedures. The exact actions taken and the time at which these actions occur depend on what systems are available and the plans for further plant operation.

However, for these events, operator actions are not required to maintain the plant in a safe and stable condition. Operator actions typical of normal operation are credited for the inadvertent actuations of equipment in response to a Condition II event.

15.2.6.1 Identification of Causes and Accident Description

...

During a plant transient, core decay heat removal is normally accomplished by the startup feedwater system if available, which is started automatically when low levels occur in either steam generator. If that system is not available, emergency core decay heat removal is provided by the PRHR heat exchanger. The PRHR heat exchanger is a C-tube heat exchanger connected, through inlet and outlet headers, to the reactor coolant system. The inlet to the heat exchanger is from the reactor coolant system hot leg, and the return is to the steam generator outlet plenum. The heat exchanger is located above the core to provide natural circulation flow when the reactor coolant pumps are not operating. The IRWST provides the heat sink for the heat exchanger. The PRHR heat exchanger, in conjunction with the passive containment cooling system, provides core cooling and maintains reactor coolant system conditions to satisfy the evaluation criteria. After the IRWST water reaches saturation (in about two and half hours), steam starts to vent to the containment atmosphere. The condensation that collects on the containment steel shell (cooled by the passive containment cooling system) returns to the IRWST, maintaining fluid level for the PRHR heat exchanger heat sink. The analysis shows that the natural circulation flow in the reactor coolant system following a loss of ac power event is sufficient to remove residual heat from the core.

The following change would be made on Sheet 6 of Table 19.59-18, "PRA-Based Insights":

... The PRHR HX, in conjunction with the IRWST, condensate return features and the PCS, can provide core cooling for at least 72 hours. After the IRWST water reaches its saturation temperature, the process of steaming to the containment initiates...	6.3.2.1.1 & 6.3.7.6

19E.4.10.2 Shutdown Temperature Evaluation

As discussed in Section 6.3.1.1.4, the passive residual heat removal heat exchanger is required to be able to cool the reactor coolant system to 420°F or below within 36 hours after shutdown following a non-LOCA event. The following summarizes a non-bounding, conservative analysis, which demonstrates the passive residual heat removal heat exchanger can meet this criterion. This analysis demonstrates that the passive systems can bring the plant to a safe, stable condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (Reference 15) criteria.

As discussed in subsections 6.3.3 and 7.4.1.1, the PRHR HX operates to reduce the RCS temperature to the specified safe shutdown condition following a non-LOCA event. An analysis of the loss of main feedwater with loss of ac power event demonstrates that the passive systems can bring the plant to this condition following postulated transients. A non-bounding, conservative analysis is represented in Figures 19E.4.10-1 through 19E.4.10-4. The progression of this event is outlined in Table 19E.4.10-1. Though some of the assumptions in this evaluation are based on nominal conditions, many of the analysis assumptions are bounding.

The performance of the PRHR HX is affected by the containment pressure. Containment pressure determines the PRHR HX heat sink (the IRWST water) temperature. The WGOTHIC containment response model described in subsection 6.2.1.1.3 was used to determine the containment pressure response to this transient, which was used as an input to the plant cooldown analysis performed with LOFTRAN. Some changes were made to the WGOTHIC model to ensure the results were conservative for the long-term safe shutdown analysis.

The PRHR HX performance is also affected by the IRWST water level when the level drops below the top of the PRHR HX tubes. The IRWST water level is affected by the heat input from the PRHR HX and by the amount of steam that leaves the IRWST and does not return to the IRWST through the IRWST gutter arrangement. The principal steam condensate losses include steam that stays in the containment atmosphere, steam that condenses on heat sinks inside containment other than the containment vessel, and dripping or splashing losses due to obstructions on the inner containment vessel wall. The WGOTHIC containment response model also provided the mass balance with respect to the steam lost to the containment atmosphere and to condensation on passive heat sinks other than the containment vessel. The WGOTHIC analysis inputs (including the mass of the heat sinks and heat transfer rates) were biased to increase steam condensate losses. The efficiency of the gutter collection system was determined separate from the WGOTHIC analysis. The resulting time-dependent condensate return rate was incorporated into the LOFTRAN computer code described in subsection 15.0.11.2 to demonstrate that the RCS could be cooled to 420°F within 36 hours.

Summarizing this transient, the loss of normal ac power (offsite and onsite) occurs, followed by the reactor trip. The PRHR HX is actuated on the low steam generator narrow range level coincident with low startup feed water flow rate signal. Eventually a safeguards actuation signal is actuated on low cold leg temperature and the CMTs are actuated.

Once actuated, at about 2,400 seconds, the CMTs operate in recirculation mode, injecting cold borated water into the RCS. In the first part of their operation, due to the injection of cold water, the CMTs operate in conjunction with the PRHR HX to reduce RCS temperature. Due to the primary system cooldown, the PRHR heat transfer capability drops below the decay heat and the RCS cooldown is essentially driven by the CMT cold injection flow. However, at about 5,000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (Figure 19.E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about 34,500 seconds, the PRHR heat transfer matches decay heat and it continues to operate to reduce the RCS temperature to below 420°F within 36 hours. As seen from Figure 19E.4.10-1, the cold leg temperature in the loop with the PRHR is reduced to 420°F within 48,600 seconds, while the core average temperature reaches 420°F within 124,400 seconds (approximately 34.6 hours).

As discussed in subsection 7.4.1.1, a timer is used to automatically actuate the automatic depressurization system if offsite and onsite power are lost for about 24 hours. This timer automates putting the open loop cooling features into service prior to draining the Class 1E dc 24-hour batteries that operate the ADS valves. At approximately 22 hours, if the plant conditions indicate that the ADS would not be needed until well after 24 hours, the operators are directed to de-energize all loads on the 24-hour batteries. This action will block actuation of the ADS and preserves the ability to align open loop cooling at a later time. Operation of the ADS in conjunction with the CMTs, accumulators, and IRWST reduces the RCS pressure and temperature to below 420°F. The ability to actuate ADS and IRWST injection provides a safety-related, backup mode of decay heat removal that is diverse to extended PRHR HX operation.

As discussed in subsection 6.3.3.2.1.1, the PRHR HX can operate in this mode for at least 72 hours to maintain RCS conditions within the applicable Chapter 15 safety evaluation criteria. In addition, the analysis supporting this section shows the PRHR HX is expected to maintain safe shutdown conditions for more than 14 days. One important consideration with regard to the duration closed-loop cooling can be maintained is the RCS leak rate. This duration of closed-loop cooing can be achieved with expected RCS leak rates. For abnormal leak rates, it may become necessary to initiate open-loop cooling earlier than 14 days.

19E.9 References

14. Not used.

Enclosure 5
Duke Energy
Levy Nuclear Plant Units 1 and 2
Part 2 and Part 4 COL Application Revisions
(30 pages including cover page)

Associated LNP COL Application Revisions:

The following revisions to the LNP COL application represent an integrated list of revisions based on revisions identified in Enclosure 5 of Serial: NPD-NRC-2014-012 and Enclosure 1 of Serial: NPD-NRC-2014-017 as well as revisions identified in this letter. These revisions will be incorporated into the next update of the LNP COLA.

COLA Part 2, FSAR

COLA Part 2, FSAR Chapter 1, Table 1.8-201, Summary of FSAR Departures from the DCD, will be revised to add additional FSAR changes to the list of FSAR Section or Subsection references for departure LNP COL 3.2-1, to read as follows:

Departure Number	Departure Description Summary	FSAR Section or Subsection
LNP DEP 3.2-1	The condensate return portion of the Passive Core Cooling System has been upgraded to add downspouts and plug fabrication holes in the Polar Crane Girder in order to maximize the return of condensate to the In-Containment Refueling Water Storage Tank and ensure long-term operation of the Passive Residual Heat Removal Heat Exchanger to meet design requirements. The following are the departures from the DCD: Table 3.2-3 (Sheet 16 of 75), Figure 3.8.2-1 (Sheet 3), Subsections 5.4.11.2 and 5.4.14.1, Chapter 6 TOC (Table of Contents, List of Figures), Subsections 6.3.1.1.1, 6.3.1.1.4, 6.3.1.2, 6.3.1.3, 6.3.2.1, 6.3.2.1.1, 6.3.2.2.7, 6.3.2.8, 6.3.3, 6.3.3.2.1.1, Figure 6.3-1 (Sheets 1 through 3), Figure 6.3-2 (Not Used), Subsection 7.4.1.1, Table 14.3-2 (Sheets 7 and 8 of 17), Subsection 15.0.13, Chapter 16 (TS Bases B3.3.3 and B3.5.4), Subsection 19E.4.10.2, Table 19E.4.10-1, Figures 19E.4.10-1 through 19E.4.10-4, and 19E.9.	Table 3.2-202, Figure 3.8-201, 5.4.11.2, 5.4.14.1, 6 TOC (List of Figures), 6.3.1.1.1, 6.3.1.1.4, 6.3.1.2, 6.3.1.3, 6.3.2.1, 6.3.2.1.1, 6.3.2.2.7, 6.3.2.8, 6.3.3, 6.3.3.2.1.1, Figure 6.3-201, 7.4.1.1, 14 TOC (list of Tables), Table 14.3-202, 15.0.13, 16 (TS Bases B3.3.3 and B3.5.4), 19 TOC (list of Tables and List of figures), 19E.4.10.2,

		Table 19E.4.10-201, Figures 19E.4.10-201 through 19E.4.10-204, 19E.9

COLA Part 2, FSAR Chapter 1, Table 1.8-201, Summary of FSAR Departures from the DCD, will be revised to add FSAR changes to the list of FSAR Section or Subsection references for departure LNP COL 6.3-1, to read as follows:

Departure Number	Departure Description Summary	FSAR Section or Subsection
LNP DEP 6.3-1	The DCD states that the PRHR HX can maintain safe shutdown conditions for non-LOCA accidents "indefinitely." A quantitative duration of greater than 14 days has been adopted based on that time being long enough to minimize the need to switch to passive feed and bleed cooling except for very unlikely or extreme hazard events. The following are the departures from the DCD: Subsection 5.4.14.1, Subsections 6.3.1.1.1, 6.3.1.2, 6.3.1.3, 6.3.2.1.1, 6.3.3.4.1, 7.4.1.1, Table 9.5.1-1 (Sheet 11), Subsection 15.2.6.1, Table 19.59-18 (Sheet 6), Subsection 19E.4.10.2	5.4.14.1, 6.3.1.1.1, 6.3.1.2, 6.3.1.3, 6.3.2.1.1, 6.3.3.4.1, 7.4.1.1, Table 9.5.1-201, 15.2.6.1, Table 19.59-202, 19E.4.10.2

COLA Part 2, FSAR Sections 5.4, 6.3, 7.4, 9.5, 14.3, Chapter 15, Chapter 16 and Chapter 19 will be revised to add the departures identified in Table 1.8-201 with a LMA of LNP DEP 3.2-1 or 6.3-1, as presented below.

1. COLA Part 2, FSAR Chapter 5, will be revised to add new Subsection 5.4.11.2, with a LMA of LNP DEP 3.2-1, to read:

   5.4.11.2 System Description

   Replace the second sentence of the second paragraph of DCD Subsection 5.4.11.2 with the following:

   The piping and instrumentation diagram for the connection between the automatic depressurization system valves and the in-containment refueling water storage tank is shown in Figure 6.3-1.

2. COLA Part 2, FSAR Chapter 5, will be revised to add new Subsection 5.4.14.1 to read:

   5.4.14.1 Design Bases

   Replace the first sentence of the first paragraph of DCD Subsection 5.4.14.1 with the following, with a LMA of LNP DEP 6.3-1:

   The passive residual heat removal heat exchanger automatically actuates to remove core decay heat for an extended period of time as discussed in Section 6.3, assuming the condensate from steam generated in the in-containment refueling water storage tank (IRWST) is returned to the tank.

   Combine the second and third paragraphs of DCD Subsection 5.4.14.1 and revise to read as follows, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1:

   The passive residual heat removal heat exchanger and the in-containment refueling water storage tank are designed to delay significant steam release to the containment for at least one hour. The passive residual heat removal heat exchanger will remove sufficient decay heat from the reactor coolant system to satisfy the applicable post-accident safety evaluation criteria detailed in Chapter 15. In addition, the passive residual heat removal heat exchanger will cool the reactor coolant system, with reactor coolant pumps operating or in the natural circulation mode, so that the reactor coolant system pressure can be lowered to reduce stress levels in the system if required. See Section 6.3 for a discussion of the capability of the passive core cooling system.

3. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.1.1, with a LMA of LNP DEP 3.2-1, to read:

   6.3.1.1.1 Emergency Core Decay Heat Removal

   Add new second and third bullets in the first paragraph of DCD Subsection 6.3.1.1.1 to read as follows:

   - The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate collection features and the passive containment cooling system, are designed to remove decay heat following a design basis event. Automatic depressurization actuation is not expected; but may occur depending on the amount of reactor coolant system leakage and when normal systems are recovered (refer to subsection 6.3.1.1.4).

   - The passive residual heat removal heat exchanger is designed to maintain acceptable reactor coolant system conditions for at least 72 hours following a non-LOCA event. The applicable post-accident safety evaluation criteria are discussed in Chapter 15. Operator action may be taken in accordance with emergency procedures to de-energize the loads on the Class 1E batteries to avoid unnecessary automatic actuation of the automatic depressurization system. Specific safe shutdown criteria are described in subsection 6.3.1.1.4.

Replace the fourth bullet (old second bullet) in the first paragraph of DCD Subsection 6.3.1.1.1 with the following:

- The passive residual heat removal heat exchanger is capable of performing its post-accident safety functions, assuming the steam generated in the in-containment refueling water storage tank is condensed on the containment vessel and returned by gravity via the in-containment refueling water storage tank condensate return gutter and downspouts.

4. Continue to revise the first paragraph of DCD Subsection 6.3.1.1.1 by deleting entirely the fifth bullet (old third bullet). Show as "(Deleted - new fifth bullet (old third bullet))" with LMAs of LNP DEP 3.2-1 and 6.3-1.

5. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.1.4, with a LMA of LNP DEP 3.2-1, to read:

   6.3.1.1.4 Safe Shutdown

   Replace the first two paragraphs of DCD Subsection 6.3.1.1.4 with the following three paragraphs, to read:

   The functional requirements for the passive core cooling system specify that the plant be brought to a stable condition using the passive residual heat removal heat exchanger for events not involving a loss of coolant. As stated in subsection 6.3.1.1.1, the passive residual heat removal heat exchanger in conjunction with the passive containment cooling system provides sufficient heat removal to satisfy the post-accident safety evaluation criteria for at least 72 hours. Additionally, the passive core cooling system, in conjunction with the passive containment cooling system, has diverse capability to establish long-term safe shutdown conditions in the reactor coolant system, eventually cooling the reactor coolant system to about 420°F in 36 hours, with or without availability of the reactor coolant pumps.

   The core makeup tanks automatically provide injection to the reactor coolant system after they are actuated on low reactor coolant temperature or low pressurizer pressure or level. The passive core cooling system can maintain stable plant conditions for a long time in this mode of operation, depending on the reactor coolant leakage and the availability of ac power sources. For example, with a technical specification leak rate of 10 gpm, stable plant conditions can be maintained for at least 10 hours. With a smaller leak a longer time is available.

   In most sequences the operators would return the plant to normal system operations and terminate passive system operation in accordance with the plant emergency operating procedures. In scenarios when ac power sources are unavailable for approximately 22 hours, the automatic depressurization system will automatically actuate. However, after initial plant cooldown following a non-LOCA event, operators will assess plant conditions and have the option to perform recovery actions to further cool and depressurize the reactor coolant system in a closed-loop mode of operation, i.e., without actuation of the automatic depressurization system. After verifying the reactor coolant system is in an acceptable, stable condition such that automatic depressurization is not needed, the operators may take action to extend passive residual heat removal heat exchanger operation by de-energizing the loads on the Class 1E dc batteries powering the protection and monitoring system actuation cabinets. After operators have taken action to extend its operation, the passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, has the capability to establish and maintain long-term safe shutdown conditions. The automatic depressurization system remains available to maintain safe shutdown conditions at a later time.

Replace the first sentence of the fifth paragraph (old fourth paragraph) of DCD Subsection 6.3.1.1.4 with the following:

The basis used to define the passive core cooling system functional requirements is derived from Section 7.4 of the Standard Review Plan.

Add a last sentence to the fifth paragraph (old fourth paragraph) of DCD Subsection 6.3.1.1.4, to read as follows:

The performance of the passive residual heat removal heat exchanger to bring the plant to 420°F in 36 hours is summarized in subsection 19E.4.10.2.

6. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.2 (new DCD Subsection 6.3.1.2), with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1, to read:

6.3.1.2 Nonsafety Design Basis

6.3.1.2.1 Post-Accident Core Decay Heat Removal

The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features and the passive containment cooling system, has the capability to maintain the reactor coolant system in the specified, long-term safe shutdown condition for 14 days in a closed-loop mode of operation. The automatic depressurization system can be manually actuated by the operators at any time during extended passive residual heat removal heat exchanger operation to initiate open-loop cooling. The operator actions necessary to achieve safe shutdown using the passive residual heat removal heat exchanger in a closed-loop mode of operation involve preventing unnecessary actuation of the automatic depressurization system as detailed in subsection 7.4.1.1.

7. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.1.3, title only, to reflect the numbering change of DCD Subsection 6.3.1.2 to 6.3.1.3, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1, to read as follows:

6.3.1.3 Power Generation Design Basis

8. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.1, with a LMA of LNP DEP 3.2-1, to read:

   6.3.2.1 Schematic Piping and Instrumentation Diagrams

   Replace the first sentence of the first paragraph of DCD Subsection 6.3.2.1 with the following:

   Figure 6.3-1 shows the piping and instrumentation drawings of the passive core cooling system.

9. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.1.1 to read:

   6.3.2.1.1 Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions

   Replace the seventh and eighth paragraphs of DCD Subsection 6.3.2.1.1 with the following, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1:

   The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, condensate return features and the passive containment cooling system, can provide core cooling for at least 72 hours. After the in-containment refueling water storage tank water reaches its saturation temperature (in several hours), the process of steaming to the containment initiates. Containment pressure will increase as steam is released from the in-containment refueling water storage tank. As the containment temperature increases, condensation begins to form on the subcooled metal and concrete surfaces inside containment. Condensation on these heat sink surfaces transfers energy to the bulk metal and concrete until they come into equilibrium with the containment atmosphere. Condensation that is not returned to the in-containment refueling water storage tank drains to the containment sump.

   Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. Most of the condensate formed on the containment vessel wall is collected in a safety-related gutter arrangement. A gutter is located near the operating deck elevation, and a downspout piping system is connected at the polar crane girder and internal stiffener, to collect steam condensate inside the containment during passive containment cooling system operation and return it to the in-containment refueling water storage tank. The gutter and downspouts normally drain to the containment sump, but when the passive residual heat removal heat exchanger actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the in-containment refueling water storage tank. Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for an extended period of time.

   Revise the first and second sentences of the ninth paragraph of DCD Subsection 6.3.2.1.1 to read as follows, with a LMA of LNP DEP 3.2-1:

The passive residual heat removal heat exchanger is used to maintain an acceptable, stable reactor coolant system condition. It transfers decay heat and sensible heat from the reactor coolant system to the in-containment refueling water storage tank, the containment atmosphere, the containment vessel, and finally to the ultimate heat sink–the atmosphere outside of containment.

Add a new tenth paragraph to DCD Subsection 6.3.2.1.1 to read as follows, with a LMA of LNP DEP 3.2-1:

The duration the passive residual heat removal heat exchanger can continue to remove decay heat is affected by the efficiency of the return of condensate to the in-containment refueling water storage tank. The in-containment refueling water storage tank water level is affected by the amount of steam that leaves the tank and does not return. Offsite or onsite ac power sources are typically recovered within a day, which would allow the operators to place active, defense-in-depth systems into service and to terminate passive system operation. If ac power is not recovered within this time frame, closed-loop cooling using the passive residual heat removal heat exchanger can be extended as described in subsection 7.4.1.1 to maintain a safe, stable condition after a design basis event.

10. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7, with a LMA of LNP DEP 3.2-1, to read:

    6.3.2.2.7 IRWST and Containment Recirculation Screens

    Replace the first paragraph of DCD Subsection 6.3.2.2.7 with the following:

    The passive core cooling system has two different sets of screens that are used to prevent debris from entering the reactor and blocking core cooling passages during a LOCA: IRWST screens and containment recirculation screens. The screens are AP1000 Equipment Class C and are designed to meet seismic Category I requirements. The structural frames attachment to the building structure, and attachment of the screen modules use the criteria of ASME Code, Section III Subsection NF. The screen modules are fabricated of sheet metal and are designed and fabricated to a manufacturer's standard. The IRWST screens and containment recirculation screens are designed to comply with applicable licensing regulations including:
    - GDC 35 of 10 CFR 50 Appendix A
    - Regulatory Guide 1.82
    - NUREG-0897

11. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7.1, with a LMA of LNP DEP 3.2-1, to read:

    6.3.2.2.7.1 General Screen Design Criteria

    Replace the first paragraph of DCD Subsection 6.3.2.2.7.1 with the following:

The IRWST screens and containment recirculation screens are designed to comply with the following criteria.

12. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.2.7.2, with a LMA of LNP DEP 3.2-1, to read:

    6.3.2.2.7.2 IRWST Screens

    Replace the third paragraph of DCD Subsection 6.3.2.2.7.2 with the following:

    During a LOCA, steam vented from the reactor coolant system condenses on the containment shell and drains down the shell to the polar crane girder or internal stiffener where it is drained via downspouts to the IRWST. Steam that condenses below the internal stiffener drains down the shell and is collected in a gutter near the operating deck elevation. It is very unlikely that debris generated by a LOCA can reach the downspouts or the gutter because of their locations. Each downspout inlet is covered with a coarse screen that prevents larger debris from entering the downspout. The gutter is covered with a trash rack which prevents larger debris from clogging the gutter or entering the IRWST through the two 4-inch drain pipes. The inorganic zinc coating applied to the inside surface of the containment shell is safety – Service Level I, and will stay in place and will not detach.

13. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.2.8, with a LMA of LNP DEP 3.2-1, to read:

    6.3.2.8 Manual Actions

    Add a new third paragraph of DCD Subsection 6.3.2.8 to read as follows:

    The operator can take action to avoid actuation of the automatic depressurization system when it is not needed. For non-LOCA events during which ac power has been lost for more than 22 hours, the protection and safety monitoring system will automatically open the automatic depressurization system valves to begin a controlled depressurization of the reactor coolant system and, eventually, containment floodup and recirculation prior to depletion of the actuation batteries. However, the operators can take action to block actuation of the automatic depressurization system should actuation be deemed unnecessary based on reactor coolant system conditions. This action allows closed loop passive residual heat removal heat exchanger operation to continue as long as acceptable reactor coolant system conditions are maintained.

    Add a new first sentence to the fourth paragraph (old third paragraph) of DCD Subsection 6.3.2.8, to read as follows:

    Section 7.4 describes the anticipated operator actions to block unnecessary automatic depressurization system actuation.

14. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3, with a LMA of LNP DEP 3.2-1, to read:

6.3.3 Performance Evaluation

Replace the seventh paragraph of DCD Subsection 6.3.3 with the following:

For non-LOCA events, the passive residual heat removal heat exchanger is actuated so that it can remove core decay heat. The passive residual heat removal heat exchanger can operate for at least 72 hours after initiation of a design basis event to satisfy Condition I, II, III and IV safety evaluation criteria described in the relevant safety analyses. Subsection 6.3.3.2.1 provides an evaluation of the duration of the passive residual heat removal heat exchanger operation using the LOFTRAN code described in subsection 15.0.11.2. In this evaluation it is assumed that the operators power down the protection and monitoring actuation cabinets in the 22 hour time frame prior to the automatic timer actuating ADS.

Add a new eighth paragraph to DCD Subsection 6.3.3, as follows:

In addition to mitigating the initiating events, the passive residual heat removal heat exchanger is capable of cooling the reactor coolant system to the specified safe shutdown condition as described in subsection 19E.4.10.2. A non-bounding, conservative estimation of the plant response during operator-initiated, extended operation of the passive residual heat removal heat exchanger is demonstrated in the shutdown temperature evaluation of subsection 19E.4.10.2. The closed-loop cooling mode allows the reactor coolant system pressure to decrease and reduces the stress in the reactor coolant system and connecting pipe to low levels. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.

Add the following as the last sentence to the tenth paragraph (old ninth paragraph) of DCD Subsection 6.3.3, as follows:

If onsite or offsite ac power has not been restored after 72 hours, the post-72 hour support actions described in subsection 1.9.5.4 maintain this mode of core cooling and provide adequate decay heat removal for an unlimited time.

Add a new eleventh paragraph to DCD Subsection 6.3.3, as follows:

The transient analyses summarized in Chapter 15 are extended long enough to demonstrate the applicable safety evaluation criteria are met. It is expected that normal systems would be available such that operators could terminate the passive safety systems and proceed with an orderly shutdown. However, as discussed in subsection 6.3.1.1.4, the passive systems are capable of bringing the plant to a safe shutdown condition and maintaining that condition.

15. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3.2.1.1 (new DCD Subsection 6.3.3.2.1.1), with a LMA of LNP DEP 3.2-1, to read:

    6.3.3.2.1.1 Loss of AC Power to the Plant Auxiliaries

    The most severe conditions resulting from a loss of ac power to the plant auxiliaries are associated with loss of offsite power with a loss of main feedwater system flow at full power. A loss of main feedwater with a loss of ac power lasting longer than a few hours presents the highest demand on passive residual heat removal heat exchanger operation. Subsection 15.2.6 provides a description of this short-term event, including criteria and analytical results.

    During most events, the passive systems would be terminated in hours. However, if normal systems are not recovered as expected, the passive residual heat removal heat exchanger removes core decay heat and maintains acceptable reactor coolant system conditions for at least 72 hours. For a non-loss of coolant accident event lasting as long as 24 hours, the automatic depressurization system will actuate if operators do not act to avoid actuation when it is not needed. For this long-term transient, it is assumed operators extend passive residual heat removal heat exchanger operation as described in subsection 7.4.1.1, such that the automatic depressurization system does not actuate.

    The loss of main feedwater with loss of ac power event is analyzed for a 72 hour period, assuming operators extend closed-loop cooling beyond the time the automatic depressurization system would be actuated by the protection and safety monitoring system. This event mirrors the loss of ac power to the plant auxiliaries event described in subsection 15.2.6, but the loss of ac power extends to 72 hours. In this event, operation of the passive residual heat removal heat exchanger continues for 72 hours and maintains acceptable reactor coolant system conditions such that the applicable Condition II safety evaluation criteria are met.

    Reactor coolant system leakage could limit closed-loop capacity. A reactor coolant system leak could produce conditions that would preclude the operators from de-energizing the loads on the Class 1E batteries, or could require the operators to re-energize the buses powered by the Class 1E batteries before 72 hours so that the automatic depressurization system valves could be actuated. When an ac power source is restored and passive core cooling system termination criteria are satisfied, the operator terminates passive core cooling system operation and initiates normal plant shutdown operations.

16. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.3.3.4.1, with a LMA of LNP DEP 6.3-1, to read:

    6.3.3.4.1 Loss of Startup Feedwater During Hot Standby, Cooldowns, and Heat-ups

Revise the last sentence of the fourth paragraph of DCD Subsection 6.3.3.4.1 to read as follows:

This allows it to function as a heat sink.

17. COLA Part 2, FSAR Section 6.3 will be revised to add a departure from DCD Figure 6.3-1 as Figure 6.3-201, as shown in Sheets 1 through 3 of Figure 6.3-201 in the attachment to this enclosure, with a LMA of LNP DEP 3.2-1. These sheets replace the figure added as Figure 6.3-201 in LNP COLA Revision 6.

18. COLA Part 2, FSAR Chapter 7, will be revised to add new Subsection 7.4.1.1, to read:

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

7.4.1.1 Safe Shutdown Using Safety-Related Systems

Revise the second sentence of the sixth paragraph of DCD Subsection 7.4.1.1 to read as follows, with a LMA of LNP DEP 6.3-1:

This prevents loss of water inventory from containment and permits extended operation of the passive residual heat removal heat exchanger and the in-containment refueling water storage tank.

Revise the last sentence of the eighth paragraph of DCD Subsection 7.4.1.1 to read as follows, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1:

The system provides core decay heat removal in this configuration without an increase in the containment water level.

Revise the ninth paragraph of DCD Subsection 7.4.1.1 to read as follows, with a LMA of LNP DEP 3.2-1:

Once the reactor coolant system and the safety systems are in this configuration, the plant is in a stable shutdown condition. The reactor coolant system temperatures and pressures continue to slowly decrease. The passive residual heat removal heat exchanger has the capacity to maintain a safe, stable reactor coolant system condition during a design basis event for at least 72 hours in a closed-loop mode of operation. A non-bounding, conservative analysis of extended operation in this mode shows the passive residual heat removal heat exchanger cools the reactor coolant system to 420°F in 36 hours.

Revise the last three sentences of the eleventh paragraph of DCD Subsection 7.4.1.1 to read as follows, with a LMA of LNP DEP 3.2-1:

The operator assessment includes consideration for a visible refueling water storage tank level, full core makeup tanks, a high and stable pressurizer level, and decreasing or stable reactor coolant system temperature. If automatic depressurization is not needed, the operator is directed to de-energize all loads on the Class 1E dc batteries. This action preserves the capability for the operator to initiate automatic depressurization at a later time based on assessment of the same parameters.

19. COLA Part 2, FSAR Section 9.5 will be revised to add a departure from DCD Table 9.5.1-1, AP1000 Fire Protection Program Compliance with BTP CMEB 9.5-1, Sheet 11 of 33, as new FSAR Table 9.5.1-201, Sheet 1, with a LMA of LNP DEP 6.3-1. This Table shall also be added to the list of tables in Chapter 9. Table 9.5.1-201, Sheet 1, is shown in the attachment to this enclosure.

20. COLA Part 2, FSAR Section 14.3 will be revised to add a departure from DCD Table 14.3-2, Design Basis Accident Analysis, Sheets 7 and 8 of 17, as new FSAR Table 14.3-202, Sheets 1 and 2, with a LMA of LNP DEP 3.2-1. This Table shall also be added to the list of tables in Chapter 14. Table 14.3-202, Sheets 1 and 2, are shown in the attachment to this enclosure.

21. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.0.13, with a LMA of LNP DEP 3.2-1, to read:

    15.0.13 Operator Actions

    Revise the first sentence of the first paragraph of DCD Subsection 15.0.13 to read as follows:

    For events where the PRHR heat exchanger is actuated, the plant automatically cools down to a safe, stable condition.

22. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.2.6.1, with a LMA of LNP DEP 6.3-1, to read:

    This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

    15.2.6.1 Identification of Causes and Accident Description

    Revise the seventh sentence of the fourth paragraph of DCD Subsection 15.2.6.1 to read as follows:

    The PRHR heat exchanger, in conjunction with the passive containment cooling system, provides core cooling and maintains reactor coolant system conditions to satisfy the evaluation criteria.

23. COLA Part 2, FSAR Section 19.59 will be revised to add a departure from DCD Table 19.59-18, PRA Based Insights, Sheet 6 of 25, as new FSAR Table 19.59-202, Sheet 1, with a LMA of LNP DEP 6.3-1. This Table shall also be added to the list of tables in Chapter 19. Table 19.59-202, Sheet 1, is shown in the attachment to this enclosure.

24. COLA Part 2, FSAR Chapter 19, Appendix 19E Shutdown Evaluation, will be revised as follows, with a LMA of LNP DEP 3.2-1:

    This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

    19E.4.10.2 Shutdown Temperature Evaluation

    Revise the first and second paragraphs of DCD Subsection 19E.4.10.2 to read as follows:

    As discussed in Section 6.3.1.1.4, the passive residual heat removal heat exchanger is required to be able to cool the reactor coolant system to 420°F or below within 36 hours after shutdown following a non-LOCA event. The following summarizes a non-bounding, conservative analysis, which demonstrates the passive residual heat removal heat exchanger can meet this criterion. This analysis demonstrates that the passive systems can bring the plant to a safe, stable condition and maintain this condition so that no transients will result in the specified acceptable fuel design limit and pressure boundary design limit being violated and that no high-energy piping failure being initiated from this condition results in 10 CFR 50.46 (Reference 15) criteria.

    As discussed in subsections 6.3.3 and 7.4.1.1, the PRHR HX operates to reduce the RCS temperature to the specified safe shutdown condition following a non-LOCA event. An analysis of the loss of main feedwater with loss of ac power event demonstrates that the passive systems can bring the plant to this condition following postulated transients. A non-bounding, conservative analysis is represented in Figures 19E.4.10-1 through 19E.4.10-4. The progression of this event is outlined in Table 19E.4.10-1. Though some of the assumptions in this evaluation are based on nominal conditions, many of the analysis assumptions are bounding.

    Add new paragraphs 3 and 4 to DCD Subsection 19E.4.10.2 to read as follows:

    The performance of the PRHR HX is affected by the containment pressure. Containment pressure determines the PRHR HX heat sink (the IRWST water) temperature. The WGOTHIC containment response model described in subsection 6.2.1.1.3 was used to determine the containment pressure response to this transient, which was used as an input to the plant cooldown analysis performed with LOFTRAN. Some changes were made to the WGOTHIC model to ensure the results were conservative for the long-term safe shutdown analysis.

The PRHR HX performance is also affected by the IRWST water level when the level drops below the top of the PRHR HX tubes. The IRWST water level is affected by the heat input from the PRHR HX and by the amount of steam that leaves the IRWST and does not return to the IRWST through the IRWST gutter arrangement. The principal steam condensate losses include steam that stays in the containment atmosphere, steam that condenses on heat sinks inside containment other than the containment vessel, and dripping or splashing losses due to obstructions on the inner containment vessel wall. The WGOTHIC containment response model also provided the mass balance with respect to the steam lost to the containment atmosphere and to condensation on passive heat sinks other than the containment vessel. The WGOTHIC analysis inputs (including the mass of the heat sinks and heat transfer rates) were biased to increase steam condensate losses. The efficiency of the gutter collection system was determined separate from the WGOTHIC analysis. The resulting time-dependent condensate return rate was incorporated into the LOFTRAN computer code described in subsection 15.0.11.2 to demonstrate that the RCS could be cooled to 420°F within 36 hours.

Revise the first sentence of the fifth paragraph (old third paragraph) of DCD Subsection 19E.4.10.2 to read as follows:

Summarizing this transient, the loss of normal ac power (offsite and onsite) occurs, followed by the reactor trip.

25. COLA Part 2, FSAR Chapter 19, Appendix 19E Shutdown Evaluation, will continue to be revised as follows, with a LMA of LNP DEP 3.2-1:

Revise paragraphs 6 and 7 (old paragraphs 4 and 5) of DCD Subsection 19E.4.10.2 to read as follows:

Once actuated, at about 2,400 seconds, the CMTs operate in recirculation mode, injecting cold borated water into the RCS. In the first part of their operation, due to the injection of cold water, the CMTs operate in conjunction with the PRHR HX to reduce RCS temperature. Due to the primary system cooldown, the PRHR heat transfer capability drops below the decay heat and the RCS cooldown is essentially driven by the CMT cold injection flow. However, at about 5,000 seconds, the CMT cooling effect decreases and the RCS starts heating up again (Figure 19.E.4.10-1). The RCS temperature increases until the PRHR HX can match decay heat. At about 34,500 seconds, the PRHR heat transfer matches decay heat and it continues to operate to reduce the RCS temperature to below 420°F within 36 hours. As seen from Figure 19E.4.10-1, the cold leg temperature in the loop with the PRHR is reduced to 420°F within 48,600 seconds, while the core average temperature reaches 420°F within 124,400 seconds (approximately 34.6 hours).

As discussed in subsection 7.4.1.1, a timer is used to automatically actuate the automatic depressurization system if offsite and onsite power are lost for about 24 hours. This timer automates putting the open loop cooling features into service prior to draining the Class 1E dc 24-hour batteries that operate the ADS valves. At approximately 22 hours, if the plant conditions indicate that the ADS would not be needed until well after 24 hours, the operators are directed to de-energize all loads on the 24-hour batteries. This action will block actuation of the ADS and preserves the ability to align open loop cooling at a later time. Operation of the ADS in conjunction with the CMTs, accumulators, and IRWST reduces the RCS pressure and temperature to below 420°F. The ability to actuate ADS and IRWST injection provides a safety-related, backup mode of decay heat removal that is diverse to extended PRHR HX operation.

26. COLA Part 2, FSAR Chapter 19, Appendix 19E, will continue to be revised as follows, with LMAs of LNP DEP 3.2-1 and LNP DEP 6.3-1:

    Add a new eighth paragraph to DCD Subsection 19E.4.10.2 to read as follows:

    As discussed in subsection 6.3.3.2.1.1, the PRHR HX can operate in this mode for at least 72 hours to maintain RCS conditions within the applicable Chapter 15 safety evaluation criteria. In addition, the analysis supporting this section shows the PRHR HX is expected to maintain safe shutdown conditions for more than 14 days. One important consideration with regard to the duration closed-loop cooling can be maintained is the RCS leak rate. This duration of closed-loop cooling can be achieved with expected RCS leak rates. For abnormal leak rates, it may become necessary to initiate open-loop cooling earlier than 14 days.

27. COLA Part 2, FSAR Section 19E.4.10 will be revised to add a departure from DCD Table 19E.4.10-1, Sequence of Events Following a Loss of AC Power Flow with Condensate from the Containment Shell Being Returned to the IRWST, as new FSAR Table 19E.4.10-201, with a LMA of LNP DEP 3.2-1. This Table shall also be added to the list of tables for Chapter 19. Table 19E.4.10-201 is shown in the attachment to this enclosure.

28. COLA Part 2, FSAR Chapter 19 will be revised to add a new Subsection 19E.9, with a LMA of LNP DEP 3.2-1, to read:

    19E.9 References

    14. Not used.

29. COLA Part 2, FSAR Section 19E will be revised to add a departure from DCD Figures 19E.4.10-1 through 19E.4.10-4 as Figures 19E.4.10-201 through 19E.4.10-204, with a LMA of LNP DEP 3.2-1. These figures shall also be added to the list of figures for Chapter 19. Figures 19E.4.10-201 through 19E.4.10-204 are shown in the attachment to this enclosure.

COLA Part 4, Technical Specifications

30. Revise LCO 11 for Part 4, TS Bases B 3.3.3, last sentence of the first paragraph, to read as follows, with a LMA of LNP DEP 3.2-1:

    The condensate is returned to the IRWST via a gutter and downspouts.

31. Revise the first two sentences of the third paragraph for Part 4, TS Bases B 3.5.4, Background, to read as follows, with a LMA of LNP DEP 3.2-1:

    In order to preserve the IRWST water for long-term PRHR HX operation, downspouts and a gutter are provided to collect and return water to the IRWST that has condensed on the inside surface of the containment shell. During normal plant operation, any water collected by the downspouts or gutter is directed to the normal containment sump.

32. Revise SR 3.5.4.7 of Part 4, TS Bases B 3.5.4, Surveillance Requirements, to read as follows, with a LMA of LNP DEP 3.2-1:

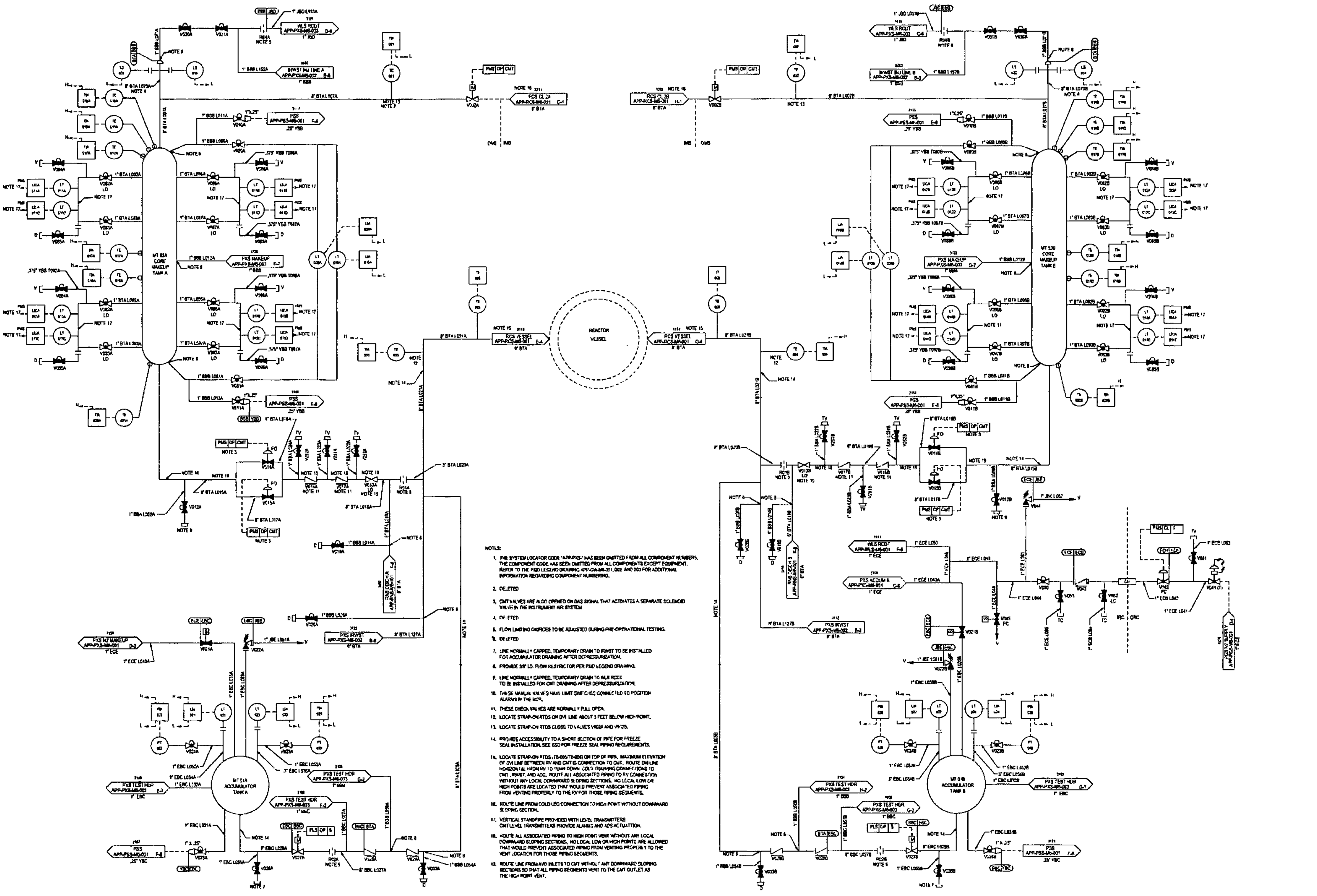
    This surveillance requires visual inspection of the IRWST gutters and downspout screens to verify that the return flow to the IRWST will not be restricted by debris. A Frequency of 24 months is adequate, since there are no known sources of debris with which the gutters or downspout screens could become restricted.

Attachments:

Figure 6.3-201, Sheets 1 through 3
Table 9.5.1-201
Table 14.3-202, Sheets 1 and 2
Table 19.59-202
Table 19E.4.10-201
Figure 19E.4.10-201
Figure 19E.4.10-202
Figure 19E.4.10-203
Figure 19E.4.10-204

Attachments to Enclosure 5
NPD-NRC-2014-021

Figure 6.3-201, Sheets 1 through 3
Table 9.5.1-201
Table 14.3-202, Sheets 1 and 2
Table 19.59-202
Table 19E.4.10-201
Figure 19E.4.10-201
Figure 19E.4.10-202
Figure 19E.4.10-203
Figure 19E.4.10-204



NOTES:

1. THE SYSTEM LOCATOR CODE "APP-PXS-" HAS BEEN OMITTED FROM ALL COMPONENT NUMBERS. THE COMPONENT CODE HAS BEEN OMITTED FROM ALL COMPONENTS EXCEPT EQUIPMENT. REFER TO THE P&ID LEGEND DRAWING APP-GW-M6-001, 002 AND 003 FOR ADDITIONAL INFORMATION REGARDING COMPONENT NUMBERING.
2. DELETED
3. CMT VALVES ARE ALSO OPENED ON DAS SIGNAL THAT ACTIVATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.
4. DELETED
5. FLOW LIMITING ORIFICES TO BE ADJUSTED DURING PRE-OPERATIONAL TESTING.
6. DELETED
7. LINE NORMALLY CAPPED, TEMPORARY DRAIN TO IRWST TO BE INSTALLED FOR ACCUMULATOR DRAINING AFTER DEPRESSURIZATION.
8. PROVIDE 3/8" I.D. FLOW RESTRICTOR PER P&ID LEGEND DRAWING.
9. LINE NORMALLY CAPPED, TEMPORARY DRAIN TO WLS RCDT TO BE INSTALLED FOR CMT DRAINING AFTER DEPRESSURIZATION.
10. THESE MANUAL VALVES HAVE LIMIT SWITCHES CONNECTED TO POSITION ALARMS IN THE MCR.
11. THESE CHECK VALVES ARE NORMALLY FULL OPEN.
12. LOCATE STRAP-ON RTDS ON DVI LINE ABOUT 3 FEET BELOW HIGH POINT.
13. LOCATE STRAP-ON RTDS CLOSE TO VALVES V002A AND V002B.
14. PROVIDE ACCESSIBILITY TO A SHORT SECTION OF PIPE FOR FREEZE SEAL INSTALLATION. SEE ESD FOR FREEZE SEAL PIPING REQUIREMENTS.
15. LOCATE STRAP-ON RTDS (TE-005/TE-006) ON TOP OF PIPE. MAXIMUM ELEVATION OF DVI LINE BETWEEN RV AND CMT IS CONNECTION TO CMT. ROUTE DVI LINE HORIZONTAL FROM RV TO TURN DOWN. COLD TRAPPING CONNECTIONS TO CMT, IRWST, AND ACC. ROUTE ALL ASSOCIATED PIPING TO RV CONNECTION WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS. NO LOCAL LOW OR HIGH POINTS ARE LOCATED THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE RV FOR THOSE PIPING SEGMENTS.
16. ROUTE LINE FROM COLD LEG CONNECTION TO HIGH POINT WITHOUT DOWNWARD SLOPING SECTION.
17. VERTICAL STANDPIPE PROVIDED WITH LEVEL TRANSMITTERS CMT LEVEL TRANSMITTERS PROVIDE ALARMS AND ADS ACTUATION.
18. ROUTE ALL ASSOCIATED PIPING TO HIGH POINT VENT WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS. NO LOCAL LOW OR HIGH POINTS ARE ALLOWED THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE VENT LOCATION FOR THOSE PIPING SEGMENTS.
19. ROUTE LINE FROM AVO INLETS TO CMT WITHOUT ANY DOWNWARD SLOPING SECTIONS SO THAT ALL PIPING SEGMENTS VENT TO THE CMT OUTLET AS THE HIGH POINT VENT.

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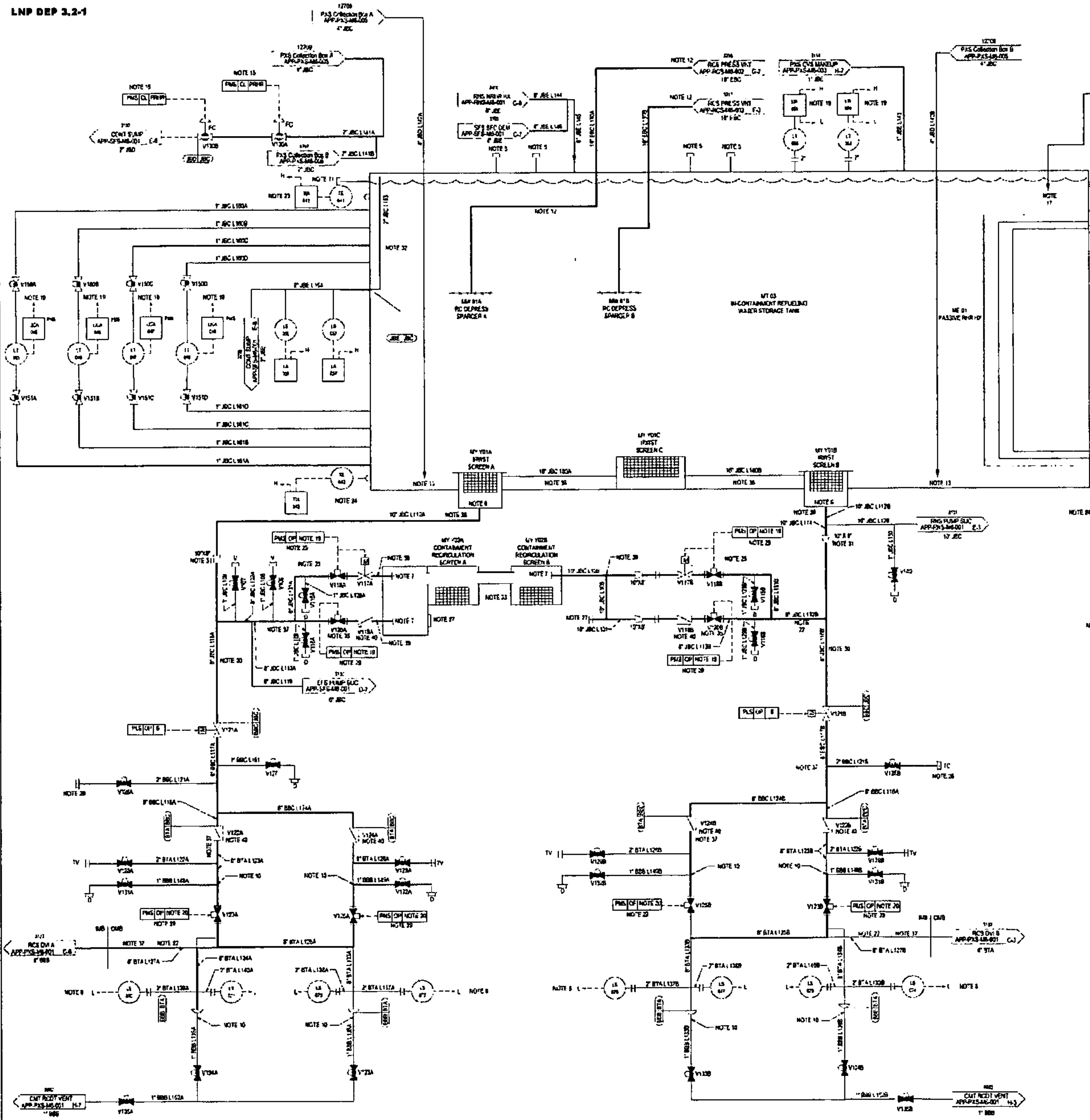
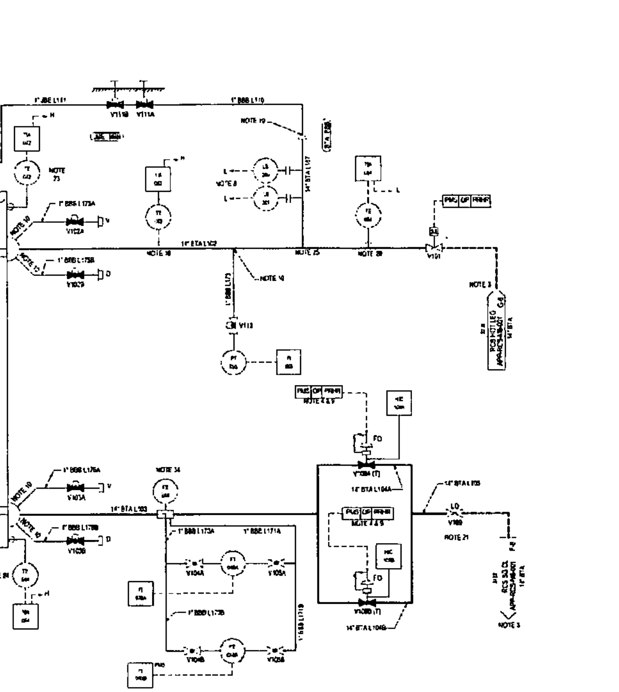
Figure 6.3-201 (Sheet 1 of 3)
Passive Core Cooling System
Piping and Instrumentation Diagram Rev. X

LNP DEP 3.2-1





NOTES:

1. THE SYSTEM LOCATOR CODE "APP-PXS-" HAS BEEN OMITTED FROM ALL COMPONENT NUMBERS. THE COMPONENT CODE HAS BEEN OMITTED FROM ALL COMPONENTS EXCEPT EQUIPMENT. REFER TO THE P&ID LEGEND DRAWING APP-GW-M6-001, 002, AND 003 FOR ADDITIONAL INFORMATION REGARDING COMPONENT NUMBERING.

3. ROUTE INLET PIPE FROM HOT LEG TO HIGH POINT PIPE STUB VENT DOWNSTREAM OF VALVE V101 WITHOUT A DOWNWARD SLOPING SECTION. ROUTE THE PRHR HX OUTLET LINE SUCH THAT IT VENTS TO EITHER THE PRHR HX OUTLET CHANNEL HEAD END OR THE SG CHANNEL HEAD END WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS IN THE DIRECTION OF EITHER VENT FROM THE LOWEST POINT BETWEEN THEM. NO LOCAL LOW OR HIGHT POINTS ARE ALLOWED IN EITHER THE INLET OR OUTLET LINES THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE LOCAL VENT PATH.

4. PRHR VALVES ARE ALSO OPENED ON DAS SIGNAL THAT ACTUATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.

5. IRWST HAS MULTIPLE VENTS. REFER TO SSD FOR LOCATION/SIZE REQUIREMENTS. EACH VENT HAS AN AIRTIGHT COVER THAT OPENS WITH A LOW DP.

6. LOCATE BOTH RCS GRAVITY INJECTION LINE CONNECTIONS SUCH THAT THEY DRAIN THE WHOLE IRWST. ENTRANCE PIPE HAS A ROUNDED CORNER WITH A RADIUS OF CURVATURE GREATER THAN OR EQUAL TO 1.5 INCH.

8. LEVEL SWITCHES SHOULD BE AT THE SAME ELEVATION.

9. PRHR SIGNAL FROM PMS OVERRIDES MODULATING SIGNAL FROM POSITIONER AND OPENS VALVE.

10. PROVIDE 3/8" I.D. FLOW RESTRICTOR PER WESTINGHOUSE P&ID LEGEND DRAWING (APP-GW-M6-001).

11. IRWST HAS MULTIPLE OVERFLOWS WHICH DISCHARGE TO THE REFUELING CAVITY LOWER PIT. REFER TO SSD FOR LOCATION/SIZE REQUIREMENTS. EACH OVERFLOW HAS AN AIRTIGHT COVER THAT OPENS WITH A LOW DP.

13. GUTTER IS LOCATED AT OPERATING DECK ELEV. TO COLLECT CONDENSATE FROM CONTAINMENT SHELL. SUMP IS PROVIDED WITH NORMALLY OPEN DRAIN TO CONTAINMENT SUMP. CLOSING V130A AND V130B CAUSES DRAIN TO OVERFLOW INTO IRWST. OVERFLOW LINE TERMINATES HORIZONTALLY 12 FEET AWAY FROM IRWST SCREEN FACE AND WITHIN 5 FEET OF THE IRWST FLOOR. GUTTER IS COVERED BY TRASH RACK.

14. THREE REDUNDANT CONTAINMENT WATER LEVEL INSTRUMENTS ARE PROVIDED COVERING CONTAINMENT LEVEL FROM BOTTOM OF REACTOR VESSEL CAVITY TO MAXIMUM FLOOD LEVEL. THE SENSORS FOR MEASURING THE CONTAINMENT FLOODUP LEVEL ARE ANALOG OUTPUT, MAGNETICALLY ACTUATED FLOAT, LEVEL SENSORS.

15. IRWST GUTTER ISOLATION VALVES ALSO CLOSE ON DAS SIGNAL THAT ACTUATES A SEPARATE SOLENOID VALVE IN THE INSTRUMENT AIR SYSTEM.

16. LOCATE STRAP-ON RTD ON PIPE APPROXIMATELY 3 FEET BELOW HIGH POINT.

17. PIPE SHOULD TERMINATE 6" BELOW NORMAL WATER LEVEL IN IRWST.

18. DELETED

19. IRWST HAS H, L, L-2, L-3 ALARMS. L-3 SETPOINT OPENS CONTAINMENT RECIRCULATION VALVES.

20. ADS STAGE 4 ACTUATION SIGNAL OR A LOW HL LEVEL SIGNAL OPENS IRWST INJECTION VALVES.

21. THIS MANUAL VALVE HAS LIMIT SWITCHES CONNECTED TO POSITION ALARMS IN MCR.

22. PROVIDE ACCESSIBILITY TO A SHORT SECTION OF PIPE FOR FREEZE SEAL INSTALLATION. SEE SSD FOR FREEZE SEAL PIPING REQUIREMENTS.

26. ROUTE 2" TEST CONNECTION LINE TO CONVENIENT LOCATION FOR HOOKUP TO A TEST CART.

27. REMOVABLE COVER TO ALLOW FOR CHECK VALVE INSERVICE TESTING WITH MECHANICAL TEST DEVICE.

28. LOCATE RTD JUST DOWNSTREAM OF VALVE V131.

29. VALVES V118A/B, V120A/B, V123A/B, V125A/B ALSO OPEN ON A DAS SIGNAL.

30. LOCATE IRWST INJECTION LINES (FROM IRWST TO RV) AND CONTAINMENT RECIRCULATION LINES (FROM CONTAINMENT TO IRWST INJECTION LINES) AT AN ELEVATION BELOW THE BOTTOM OF IRWST.

31. EXTEND 10" PIPE FROM IRWST SUMP INTO PXS VALVE ROOM. CHANGE TO 8" AS SOON AS ENTERING PXS ROOM.

32. TANK OVERFLOW CHANGE SAFETY CLASSIFICATION AT TANK PENETRATION. CONNECT WATER SENSORS IN BOTTOM OF HORIZONTAL PIPE RUN.

33. TWO SEPARATE CONTAINMENT RECIRCULATION SCREENS PROVIDED. THE TWO SCREENS ARE INTERCONNECTED BY A LINE WITH A MINIMUM FLOW AREA OF 1 SQUARE FOOT.

34. FLOW METER ELEMENT INCLUDES INTEGRAL 0.375" FLOW RESTRICTOR FOR CLASS A TO B TRANSITION SIMILAR TO P&ID LEGEND SHEET 2 NOTE 7.

35. VALVE V128A/B HAS A CLASS 2500 RATING. THESE VALVES ARE EMPLOYED TO ADD VALVE DESIGN DIVERSITY TO THE CONTAINMENT RECIRCULATION LINE.

36. ENDS OF IRWST SCREEN CROSS-CONNECT PIPE HAVE ROUNDED CORNERS WITH A RADIUS OF CURVATURE GREATER THEN OR EQUAL TO 2.25".

37. ROUTE ALL ASSOCIATED PIPING TO HIGH POINT VENT WITHOUT ANY LOCAL DOWNWARD SLOPING SECTIONS. NO LOCAL LOW OR HIGH POINTS ARE ALLOWED THAT WOULD PREVENT ASSOCIATED PIPING FROM VENTING PROPERLY TO THE VENT LOCATIONS FOR THOSE PIPING SEGMENTS.

38. ROUTE LINE FROM REDUCER TO IRWST SCREEN INLET WITHOUT A DOWNWARD SLOPING SECTION. PIPING SEGMENT MUST VENT TO IRWST OUTLET AS THE HIGH POINT.

39. ROUTE LINE FROM CONTAINMENT RECIRCULATION SCREEN TO PARALLEL SQUIB VALVES DOWNHILL WITHOUT ANY SLOPING SECTIONS SO THAT ALL PIPING SEGMENT CAN VENT TO CONTAINMENT THROUGH THE RECIRCULATION SCREENS.

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Figure 6.3-201 (Sheet 2 of 3)
Passive Core Cooling System
Piping and Instrumentation Diagram

Rev. X

LNP DEP 3.2-1

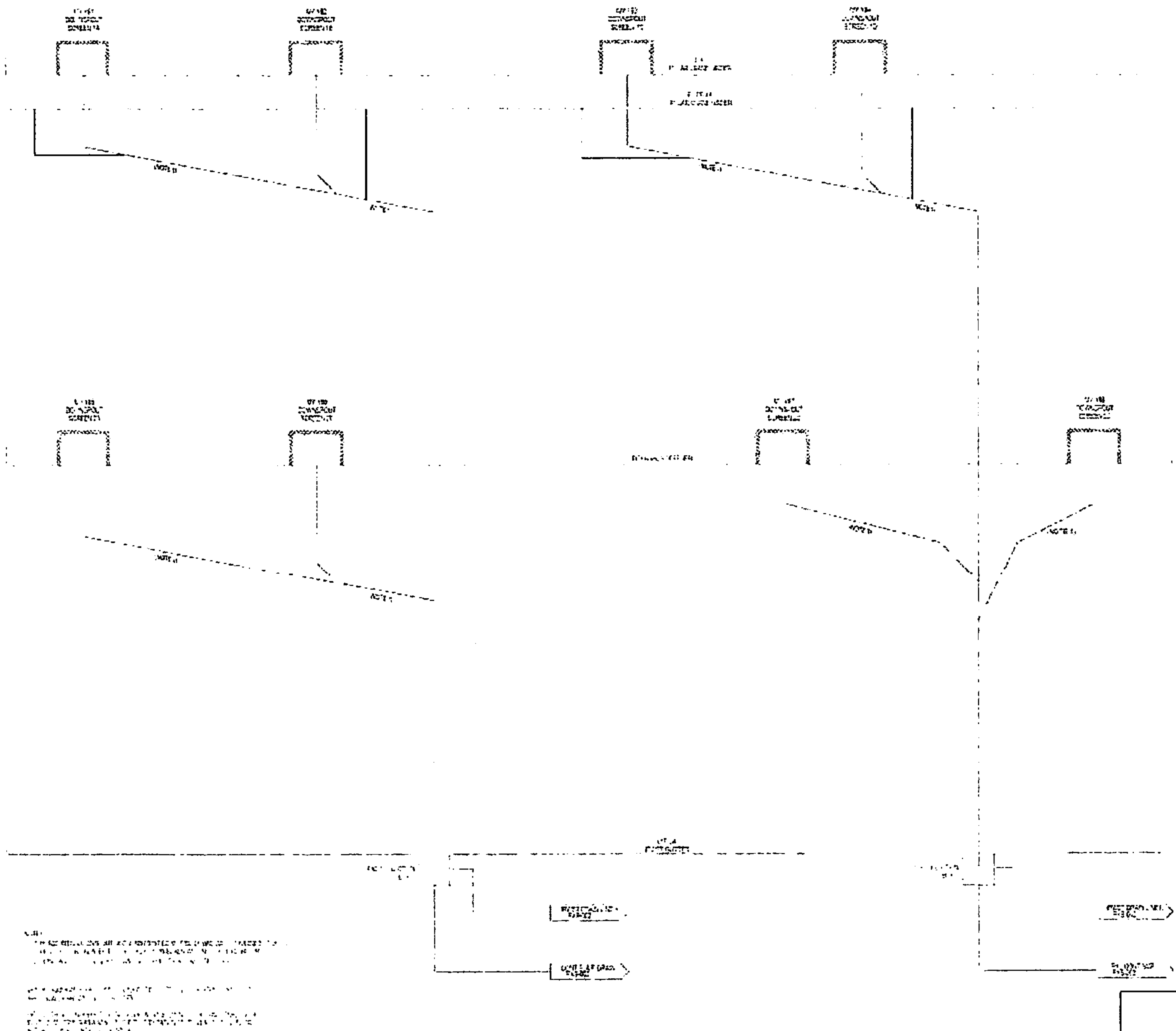

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Figure 6.3-201 (Sheet 3 of 3)
Passive Core Cooling System
Piping and Instrumentation Diagram
Rev. X

LNP DEP 6.3-1	Table 9.5.1-201		
AP1000 FIRE PROTECTION PROGRAM COMPLIANCE WITH BTP CMEB 9.5-1			
BTP CMEB 9.5-1 Guideline	**Paragraph**	**Comp[(1)]**	**Remarks**
Safe Shutdown Capability			
72. Fire damage should be limited so that one train of systems necessary to achieve and maintain hot shutdown conditions from either the main control room or emergency control station is free of fire damage.	C.5.b(1)	C	
73. Fire damage should be limited so that systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station can be repaired within 72 hours.	C.5.b (1)	AC	Safe shutdown following a fire is defined for the AP1000 plant as the ability to achieve and maintain the reactor coolant system (RCS) temperature below 215.6°C (420°F) without uncontrolled venting of the primary coolant from the RCS. This is a departure from the criteria applied to the evolutionary plant designs, and the existing plants where safe shutdown for fires applies to both hot and cold shutdown capability. With expected RCS leakage, the AP1000 plant can maintain safe shutdown conditions for at least 14 days. Therefore, repairs to systems necessary to reach cold shutdown need not be completed within 72 hours.
74. Separation requirements for verifying that one train of systems necessary to achieve and maintain hot shutdown is free of fire damage.	C.5.b (2)	C	

LNP DEP 3.2-1	Table 14.3-202 (Sheet 1 of 2) **DESIGN BASIS ACCIDENT ANALYSIS**	
Reference	**Design Feature**	**Value**
Section 6.3.6.1.3	The bottom of the in-containment refueling water storage tank is located above the direct vessel injection nozzle centerline (ft).	≥ 3.4
Section 6.3.6.1.3	The pH baskets are located below plant elevation 107′ 2″.	
Figure 6.3-1	The passive core cooling system has two direct vessel injection lines.	
Table 6.3-2	The passive core cooling system has two core makeup tanks, each with a minimum required volume (ft^3).	2500
Table 6.3-2	The passive core cooling system has two accumulators, each with a minimum required volume (ft^3)	2,000
Table 6.3-2	The passive core cooling system has an in-containment refueling water storage tank with a minimum required water volume (ft^3)	73,900
Section 6.3.2.2.3	The containment floodup volume for a LOCA in PXS room B has a maximum volume (ft^3) (excluding the IRWST) below a containment elevation of 108 feet.	73,500
Table 6.3-2	Each sparger has a minimum discharge flow area (in^2).	≥ 274
Table 6.3-2	The passive core cooling system has two pH adjustment baskets each with a minimum required volume (ft^3).	280
Section 14.2.9.1.3f	The passive residual heat removal heat exchanger minimum natural circulation heat transfer rate (Btu/hr) - With 520°F hot leg and 80°F IRWST - With 420°F hot leg and 80°F IRWST	 ≥ 1.78 E+08 ≥ 1.11 E+08
Section 6.3.6.1.3	The centerline of the HX's upper channel head is located above the HL centerline (ft).	≥ 26.3
Figure 6.3-1	The CMT level sensors (PXS-11A/B/C/D, -12A/B/C/D, -13A/B/C/D, and -14A/B/C/D) upper level tap centerlines are located below the centerline of the upper level tap connection to the CMTs (in).	1″ ± 1″
Figure 6.3-1	The CMT inlet lines (cold leg to high point) have no downward sloping sections.	
Figure 6.3-1	The maximum elevation of the CMT injection lines between the connection to the CMT and the reactor vessel is the connection to the CMTs.	
Figure 6.3-1	The PRHR inlet line (hot leg to high point) has no downward sloping sections.	

LNP DEP 3.2-1 Table 14.3-202 (Sheet 2 of 2)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Figure 6.3-1	The maximum elevation of the IRWST injection lines (from the connection to the IRWST to the reactor vessel) and the containment recirculation lines (from the containment to the IRWST injection lines) is less than the bottom inside surface of the IRWST.	
Figure 6.3-1	The maximum elevation of the PRHR outlet line (from the PRHR to the SG) is less than the PRHR lower channel head top inside surface.	
Section 7.1.2.10	Isolation devices are used to maintain the electrical independence of divisions and to see that no interaction occurs between nonsafety-related systems and the safety-related system. Isolation devices serve to prevent credible faults in circuit from propagating to another circuit.	
Section 7.1.4.2	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire, flooding, explosions, missiles, electrical faults and pipe whip.	
Section 7.1.2	The flexibility of the protection and safety monitoring system enables physical separation of redundant divisions.	
Section 7.2.2.2.1	The protection and safety monitoring system initiates a reactor trip whenever a condition monitored by the system reaches a preset level.	
Section 7.2.2.2.8	The reactor is tripped by actuating one of two manual reactor trip controls from the main control room.	
Section 7.3.1.2.2	The in-containment refueling water storage tank is aligned for injection upon actuation of the fourth stage automatic depressurization system via the protection and safety monitoring system.	
Section 7.3.1.2.3	The core makeup tanks are aligned for operation on a safeguards actuation signal or on a low-2 pressurizer level signal via the protection and safety monitoring system.	
Section 7.3.1.2.4	The fourth stage valves of the automatic depressurization system receive a signal to open upon the coincidence of a low-2 core makeup tank water level in either core makeup tank and low reactor coolant system pressure following a preset time delay after the third stage depressurization valves receive a signal to open via the protection and safety monitoring system.	

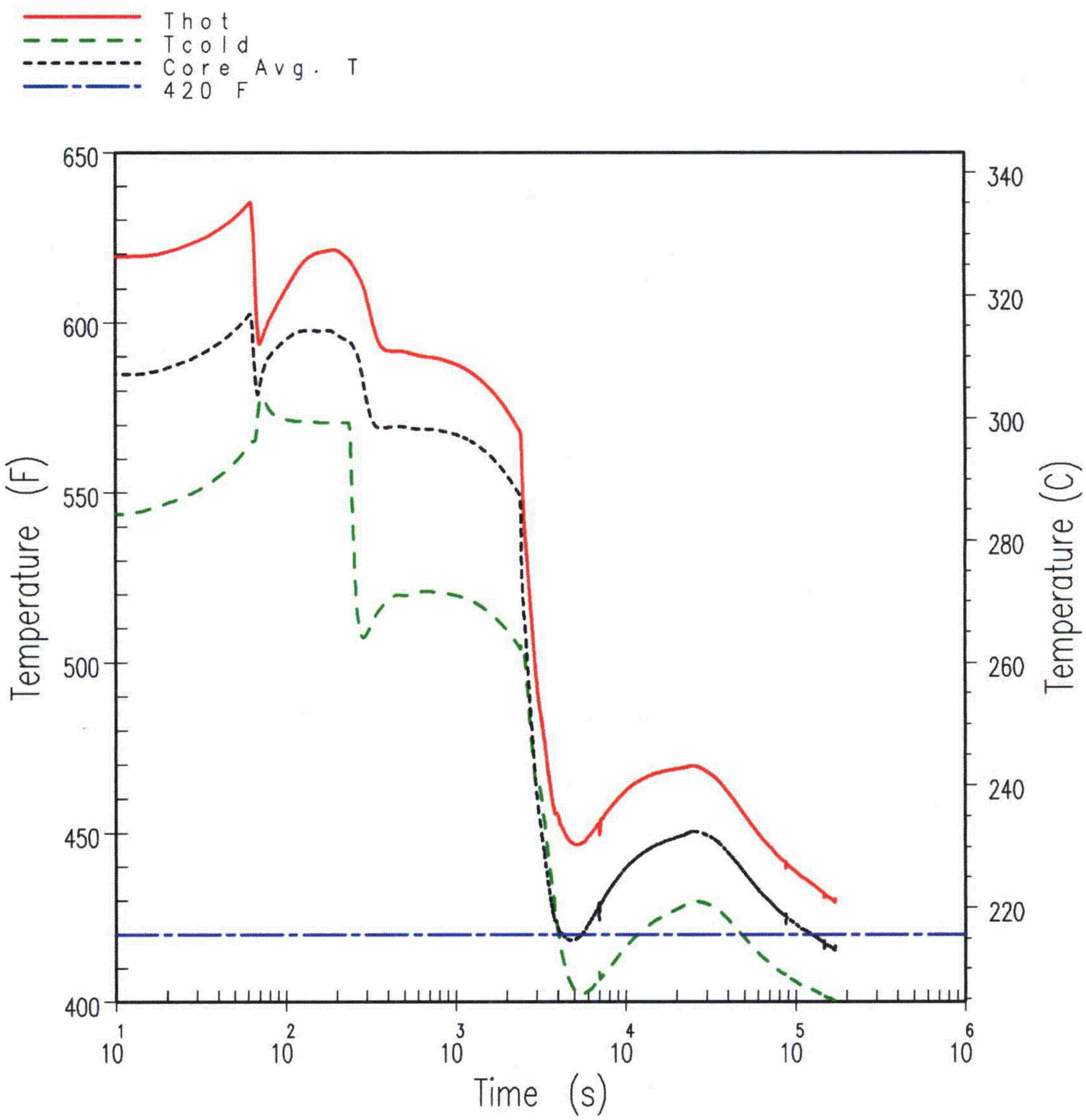
LNP DEP 6.3-1	Table 19.59-202 **AP1000 PRA-BASED INSIGHTS**
Insight	**Disposition**
1e. (cont.)	
Long-term cooling of PRHR will result in steaming to the containment. The steam will normally condense on the containment shell and return to the IRWST by safety-related features. Connections are provided to IRWST from the spent fuel system (SFS) and chemical and volume control system (CVS) to extend PRHR operation. A safety-related makeup connection is also provided from outside the containment through the normal residual heat removal system (RNS) to the IRWST.	6.3.1 & system drawings
Capability exists and guidance is provided for the control room operator to identify a leak in the PRHR HX of 500 gpd. This limit is based on the assumption that a single crack leaking this amount would not lead to a PRHR HX tube rupture under the stress conditions involving the pressure and temperature gradients expected during design basis accidents, which the PRHR HX is designed to mitigate.	6.3.3 & 16.1
The positions of the inlet and outlet PRHR valves are indicated and alarmed in the control room.	6.3.7
PRHR air-operated valves are stroke-tested quarterly. The PRHR HX is tested to detect system performance degradation every 10 years.	3.9.6
PRHR is required by Technical Specifications to be available from Modes 1 through 5 with RCS pressure boundary intact.	16.1
The PRHR HX, in conjunction with the IRWST, condensate return features and the PCS, can provide core cooling for at least 72 hours. After the IRWST water reaches its saturation temperature, the process of steaming to the containment initiates. Condensation occurs on the steel containment vessel, and the condensate is collected in a safety-related gutter arrangement, which returns the condensate to the IRWST. The gutter normally drains to the containment sump, but when the PRHR HX actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the IRWST. The following design features provide proper re-alignment for the gutter system valves to direct water to the IRWST:	6.3.2.1.1 & 6.3.7.6
- IRWST gutter and its drain isolation valves are safety-related	
- These isolation valves are designed to fail closed on loss of compressed air, loss of Class 1E dc power, or loss of the PMS signal	
- These isolation valves are actuated automatically by PMS and DAS.	7.3.1.2.7
The PRHR subsystem provides a safety-related means of removing decay heat following loss of RNS cooling during shutdown conditions with the RCS intact.	16.1

LNP DEP 3.2-1 Table 19E.4.10-201

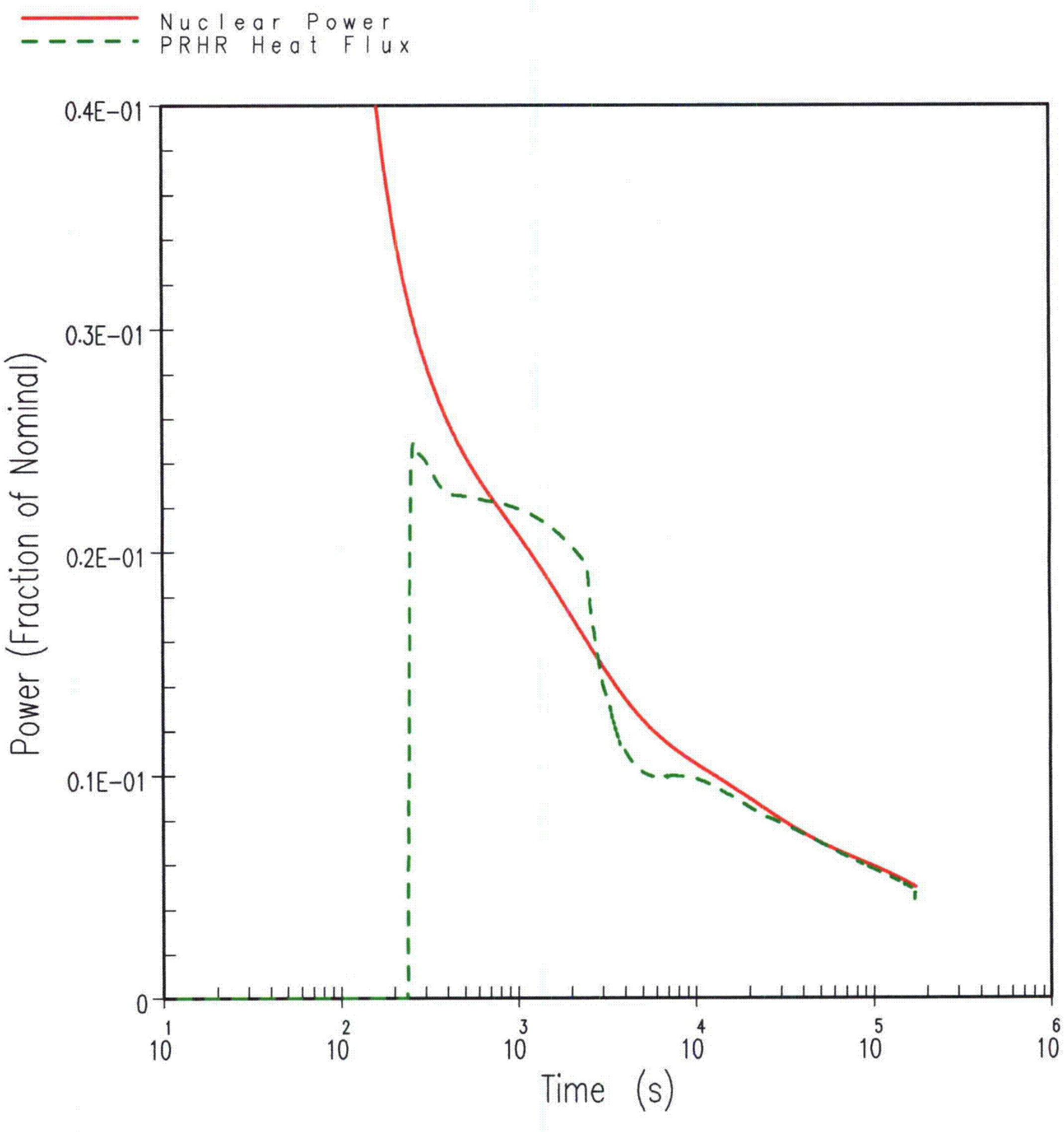
SEQUENCE OF EVENTS FOLLOWING A LOSS OF AC POWER FLOW WITH CONDENSATE FROM THE CONTAINMENT SHELL BEING RETURNED TO THE IRWST

Event	Time (seconds)
Feedwater is Lost	10.0
Low Steam Generator Water Level (Narrow-Range) Reactor Trip Setpoint Reached	≤ 60
Rods Begin to Drop	≤ 61
Low Steam Generator Water Level (Wide-Range) Reached	≤ 230
PRHR HX Actuation on Low Steam Generator Water Level (Narrow-Range Coincident with Low Startup Feedwater Flow)	≤ 240
Low T_{cold} Setpoint Reached	≤ 2400
Steam Line Isolation on Low T_{cold} Signal	≤ 2400
CMTs Actuated on Low T_{cold} Signal	≤ 2400
IRWST Reaches Saturation Temperature	≤ 15,500
Heat Extracted by PRHR HX Matches Core Decay Heat	≤ 34,500
CMTs Stop Recirculating	--
Cold Leg Temperature Reaches 420°F (loop with PRHR)	≤ 48.600
Core Average Temperature Reaches 420°F	≤ 124,400

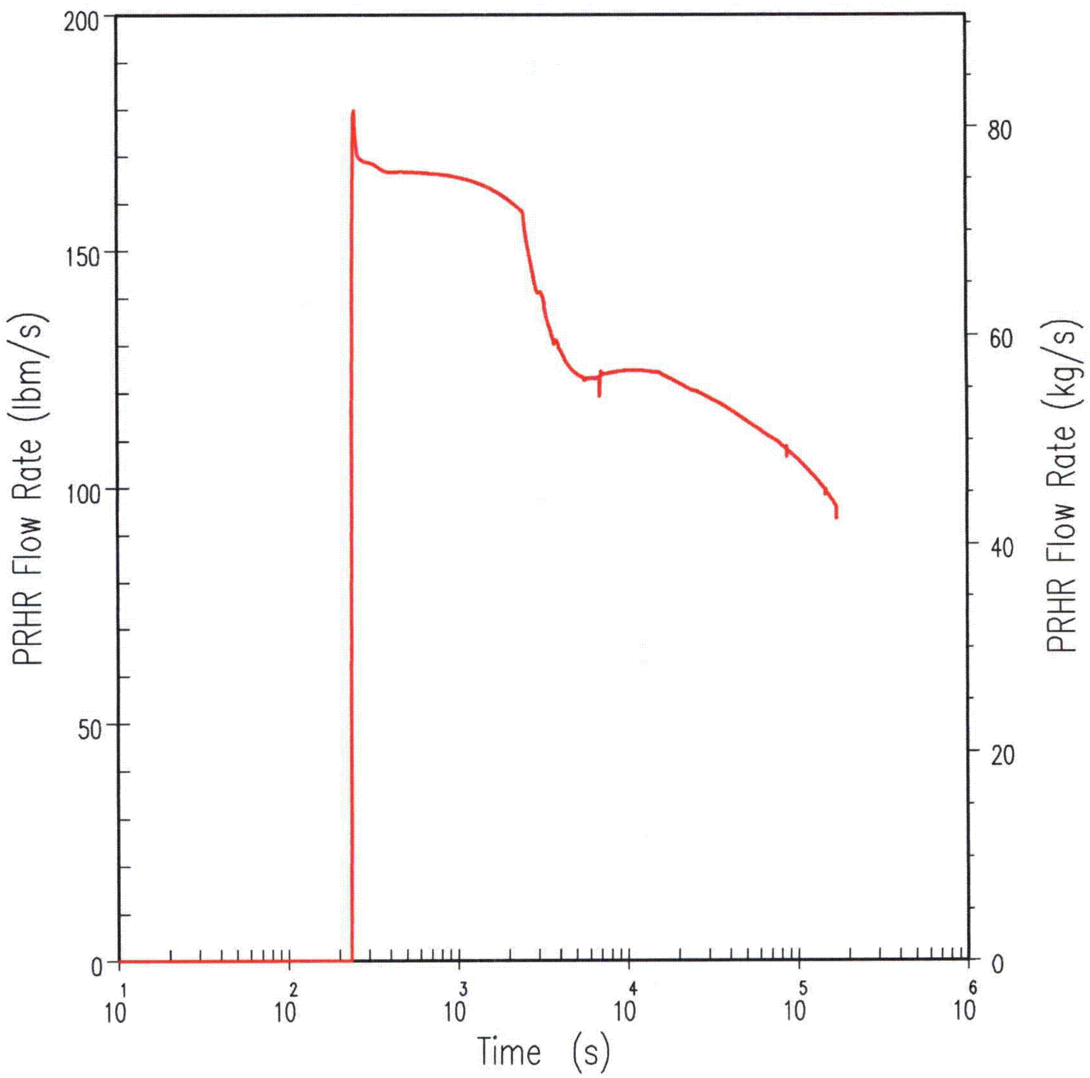
LNP DEP 3.2-1



Duke Energy Florida Levy Nuclear Plant Units 1 and 2 Part 2, Final Safety Analysis Report
Figure 19E.4.10-201 **Shutdown Temperature Evaluation, RCS Temperature** Rev. X



Duke Energy Florida **Levy Nuclear Plant** **Units 1 and 2** **Part 2, Final Safety Analysis Report**
Figure 19E.4.10-202 **Shutdown Temperature Evaluation,** **PRHR Heat Transfer** Rev. X



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Levy Nuclear Plant
Units 1 and 2
Part 2, Final Safety Analysis Report

Figure 19E.4.10-203
Shutdown Temperature Evaluation, PRHR Flow Rate

Rev. X

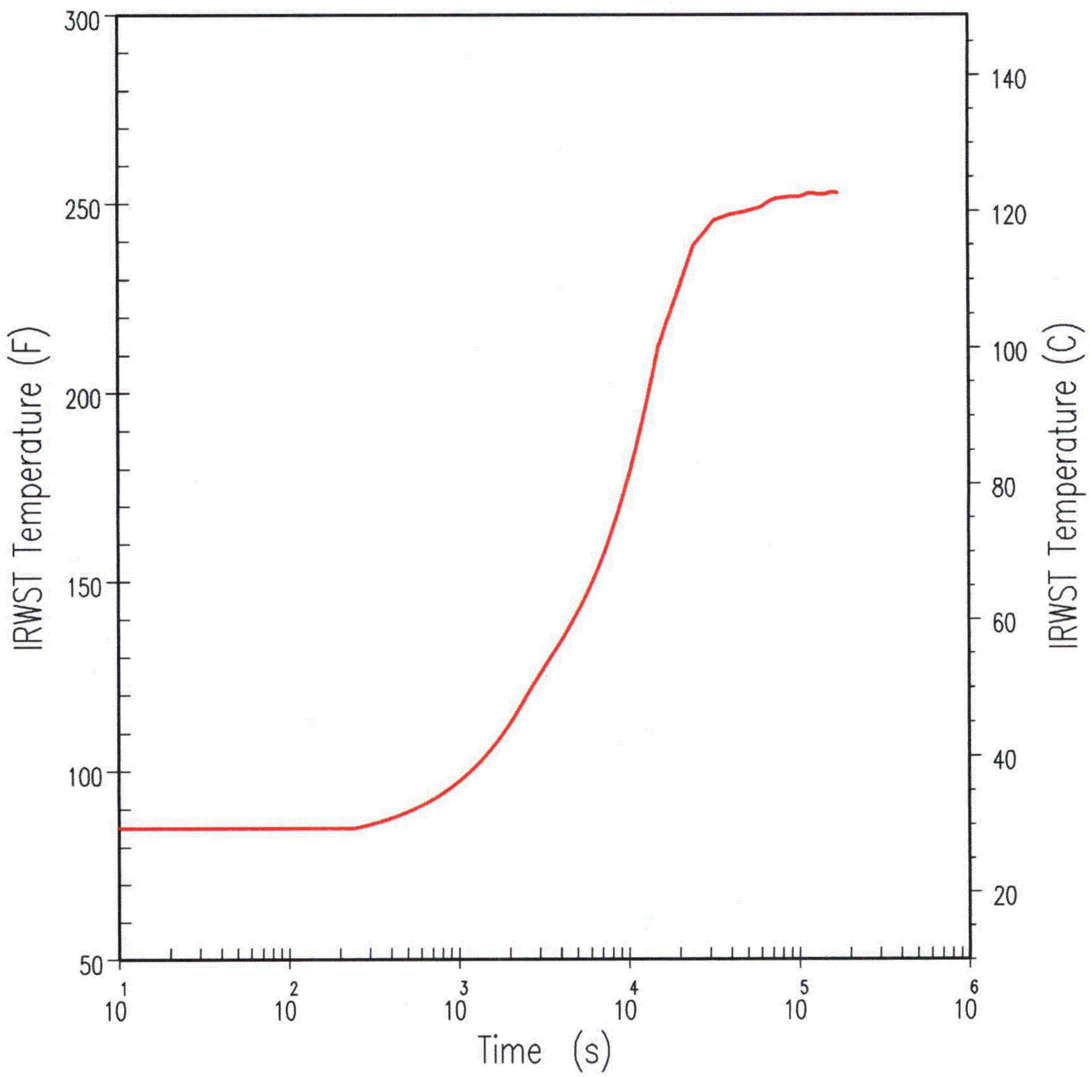

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Figure 19E.4.10-204
Shutdown Temperature Evaluation, IRWST Heatup

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